Staff Reports To The President's Commission On

THE ACCIDENT AT THREE MILE ISLAND

Reports Of The Technical Assessment Task Force, Vol. IV

REPORTS OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

QUALITY ASSURANCE

CONDENSATE POLISHING SYSTEM

CLOSED EMERGENCY FEEDWATER VALVES

PILOT-OPERATED RELIEF VALVE DESIGN AND PERFORMANCE

CONTAINMENT: TRANSPORT OF RADIOACTIVITY FROM
THE TMI-2 CORE TO THE ENVIRONS

IODINE FILTER PERFORMANCE

RECOVERY: TMI-2 CLEANUP AND DECONTAMINATION

VOLUME IV

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REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

QUALITY ASSURANCE

BY

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October, 1979
Washington, D. C.

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I. SUMMARY

A review of the independent assessment program for nuclear power plants as defined by the Nuclear Regulatory Commission (NRC) Quality Assurance, Reliability, and System Safety requirements and regulations was conducted by a team of commission staff and consultants. This team reviewed, examined, and assessed the regulations, organizations, procedures and practices involved in those NRC and utility overview activities meant to assure the safe operation of a nuclear power plant.

Requirements for these overview activities are contained in Title 10, Code of Federal Regulations, Chapter 50 (10 CFR 50), primarily Appendix B, Quality Assurance, for the utility; in the Standard Review Plan for the design review process conducted by the Office of Nuclear Reactor Regulation; and in the Inspection and Enforcement Manual for the audit program conducted by the NRC Office of Inspection and Enforcement. The staff found that the regulations and overall review process applies only to a portion of the plant defined as "safety-related" and do not fully utilize the rigorous safety analysis and reliability engineering techniques currently applied to other safety critical programs or industries. The review also indicates that the management structures that have evolved as a result of this narrow definition of NRC responsibility do not provide for an independent assessment of many critical functions and operations.

This narrow approach by NRC is reflected in a similar approach taken by the utility company responsible for the Three Mile Island Unit 2 (TMI-2) operation as it performed to just satisfy the NRC basic requirements.

The lack of independent assessment on critical activities and the narrow approach to safety taken by NRC and the utility are shown to have contributed significantly to the accident at TMI.

The following summarizes the major findings of this report:

1. The NRC organization, procedures, and practices, as now constituted, do not provide for the combined management, engineering,and assurance review of utility performance necessary to minimize the probability of equipment and operator failures and necessary to ensure the safe operation of the nuclear power plant.

2. A lack of an independent on-site quality assurance or safety assessment of plant operations and of equipment not considered "safety-related" contributed significantly to the accident at TMI.

3. There was a lack of detailed safety and failure modes analysis on all plant systems necessary to ensure the reliability and safety of the facility.

4. Systems engineering, interactions between systems and the interaction between the equipment and its operators, has not generally been considered in the NRC overview process.

5. A comprehensive, nonconformance, problem reporting, failure analysis, corrective action system applicable to all systems and operations that affect plant safety and reliability does not exist. The current license event report (LER) system also does not assure adequate dissemination and utilization of useful failure data throughout the industry.

6. Current utility and NRC practices do not assure proper preparation review and execution of operating and maintenance procedures.

7. NRC has a very limited view of changes made to nuclear power plant configuration. Utility control of safety-related equipment changes appears adequate; control of non-safety-related equipment configuration is inadequate.

8. Full use is not being made of management, engineering, safety, reliability, and quality assurance practices which are in use in other industries where safety and reliability are critical concerns.

II. INTRODUCTION

A. PURPOSE OF REPORT

This report presents and discusses the results of a review and assessment by the Commission staff of the regulations, organizations, functions, and practices of the NRC and TMI utility intended to provide management with an adequate independent assessment and overview of the performance of the nuclear power plant.

The review which was limited in scope, included the major elements of the NRC, one of its regional offices, and many of the activities of the utility company responsible for the operation of one nuclear power plant, TMI-2. During the review, weaknesses in the NRC and the utility company's independent assessment and overview process were identified. These are presented and described in this report. The remainder of this introduction describes the classical roles of major assessment organizational elements, what some of the basic terms used mean, the documented requirements and roles of the NRC and utility organizations at the time of the accident, and provides a general description of the scope of the investigation reported herein.

B. CLASSICAL ROLES

For a number of years, industry and government organizations involved with designing, building, and operating complex, dangerous systems have developed management techniques and organizational structures that consisted of:

1. The doer -- the manager, the engineer, the technician, the writer -- those that direct, design, build, prepare procedures, operate, train, and repair.

2. The checker -- quality assurance engineers, inspectors, reliability engineers, and safety engineers. The checkers are an independent assessment activity that work to prevent failures and the consequences of failures by assuring application of proper design techniques, manufacturing techniques, procedures, and training. Part of their work is to alert management to conditions of significant risk of failures or risk of unsafe conditions that remain. In no way are the checkers to detract from the responsibility of the doer to satisfactorily accomplish his assigned task.

3. The auditor -- usually a small group of independent checkers who assure that (1) and (2) are conducting their activities in accordance with plans and procedures established by management.

Experience has taught that such a structure can be effective, but to be effective, the independent assessment activities of the checkers and the auditors must have access to senior management and senior management must be receptive to the information provided by these groups and act on it to acheive the desired results.

The checkers are often organized into functional groups entitled quality assurance, reliability engineering, and safety engineering. These titles are not familiar to many people and even within the industry there are different opinions as to what each of these functions consist of and what each does. In simplistic terms they can be defined as follows:

- Quality Assurance: That discipline responsible for assuring hardware is built and operated in accordance with established drawing and procedures.

- Reliability: The discipline that assures hardware will do what it is required to do. This usually includes analyzing how hardware or people can fail, analyzing why it or they failed, and when it or they did fail.

- Safety: The discipline that evaluates the ability of the system of hardware and procedures to cope with known or potential hazards using safety practices and standards to assure that risks are recognized and reduced to acceptable levels.

Together, these elements are sometimes referred to as "product assurance," or assurance functions. They are closely interrelated and as will be shown by this staff study, a weakness in any of the three has a significant effect on the safe, reliable operation of a nuclear power plant.

C. NRC REQUIREMENTS

NRC requirements for the nuclear power industry are contained in Title 10 of the Code of Federal Regulations (CFR). The primary chapters that regulate the activities and functions covered by this paper are chapter 21 and chapter 50 and its appendices. 10 CFR 50 regulates the design, construction, and operation of the plant and chapter 21 concerns the requirements for reporting abnormal occurrences by the utility. Abnormal occurrences include certain hardware failure conditions that cause operation of the facility outside of prescribed limits, and other significant events.

10 CFR 50, Appendix B contains the basic quality assurance requirements that apply to the plant from design through operation. There are no formal "reliability" or "safety" chapters or appendices as such, but the basic code contains many sections which speak to functions that are usually considered to be reliability and safety functions. The code is very much a design requirement or engineering standard for the plant and also provides detail requirements for training and licensing of operators, radiation protection, emergency preparedness, etc. The code is much like a specification for a product the government is going to buy, but in this case, if the utility provides the product as specified, his "payment" is a license to build or operate the nuclear power plant.

D. ORGANIZATIONS INVOLVED

NRC

Prior to 1975, the responsibility for overseeing and regulating the nuclear power plant industry rested with the Atomic Energy Commission (AEC) which also had responsibility for the research and development (R&D) of nuclear power. Because of a possibility of conflict of interest between these two functions, the regulatory function was split away from R&D which was assigned to the Energy Research and Development Agency, now the U.S. Department of Energy (DOE). NRC was formed to provide the regulatory function and although it inherited and continued many of the regulations, practices, and procedures of the AEC, it had to develop many of its own, particularly since Congress had dictated part of its organization. During the period just after its formation, the Inspection and Enforcement Manual and the Standard Review Plan were developed. These two documents are critical to the utility overview process; the first defines in minute detail the role of the regional inspector in reviewing the utility, and the second defines exactly how the various NRC organizations are to review the utility's proposed design, plans, and procedures to assure the plant meets the various requirements of Title 10 of the Code of Federal Regulations.

Although using the simplistic terms just defined, one might consider NRC to be a "product assurance" agency, the organizations within NRC involved in the direct overview of the utility have many of the characteristics of a government program office buying a product. The Division of Project Management (DPM) and the Division of Operating Reactors (DOR) include the project managers, generally a coordinating function. The Division of System Safety (DSS) is the engineering function and the Office of Inspection and Enforcement (I&E) with its regional offices are the inspectors. There are other parts of NRC -- the Office of Standards Development generates standards to be utilized by the other groups, and there is some research conducted relative to the safety aspects of nuclear power by the Office of Nuclear Regulatory Research. In the case of NRC, therefore, DPM, DOR, and DSS are the "doing" function and (I&E) is the checker.

The Utility

Metropolitan Edison Company (Met Ed) is one part of a larger corporation, General Public Utilities Corporation (GPU). GPU has two other companies that build and operate nuclear power plants and a company -- General Public Utilities Service Corporation (GPUSC) -- that provides engineering and construction support to all three utilities. Although Met Ed applied for and received the licenses to construct and operate TMI-2, GPUSC actually directed the design, construction, and startup of the facility.

During 1978 the facility was undergoing startup and checkout. As each system was accepted, it was turned over to Met Ed to operate. By the end of 1978, TMI-2 was considered commercial and Met Ed had full responsibility.

Met Ed itself has two organizations concerned with TMI; the TMI site organization and the corporate office located in Reading, Pa., about 80 miles from TMI. The corporate offices of Engineering, Operations, Maintenance, and Quality Assurance have responsibility for two fossil fuel plants in addition to TMI-1 and TMI-2. The TMI site organization is primarily composed of engineering, operations, and maintenance personnel with other support groups such as training, security, and a small quality control group of inspectors. As with NRC, there is no identified reliability or safety function, but the corporate quality group has an audit group that meets NRC requirements.

E. SCOPE OF INVESTIGATION

The investigation of the quality assurance, reliability, and safety activities was initiated primarily to evaluate the NRC and utility requirements, procedures and practices associated with nonconformances, procedures, and change control. These had been identified early in the investigation by the Commission staff as possible contributors to the accident. As this review was being conducted, it became apparent that a broader analysis of the requirements, organizations, procedures, and practices of the NRC and utility as related to the independent assessment functions of quality assurance, reliability, and safety was required.

Of particular interest was whether the NRC and utility management had and used these functions in conjunction with its normal management and engineering overview to identify weaknesses in both the hardware and organizations involved in the nuclear power plant process. In short, did management know what was going on, and if not, did they have a system to tell them they did not know?

The investigation was conducted by a small team of technical staff and consultants. Visits were made to TMI, the Met Ed corporate offices in Reading, Pa., the NRC Region I offices, and to various NRC organizations in Bethesda, Md. The primary emphasis was on the operating reactor up to the time of the accident, although the roles, responsibilities and methods of the organizations involved in the construction and startup had to be understood to evaluate their relationship to, and possible effect on, the operating system. No detailed review of the design and construction phase was made; however, a brief review of Region I inspection reports, utility response, and corrective action taken was conducted to understand the depth and scope of the AEC overview in the 1970-74 time period.

Since only "quality assurance" (QA) appears in the regulations, the QA program for operating reactors was reviewed in some depth, but a detail review of all parts of the QA program was not possible. A detailed review of nonconformances, procedures, and change control was made, however, and it was generally felt that strengths or weakness in these areas would be typical of the overall quality program.

Discussions were held with supervisors, managers, or individuals from each utility and NRC organization involved in the overview process. This included NRC project management, engineering, inspection, and standards personnel, and the utility organizations, both at Reading, Pa., and TMI, involved with engineering, operations, maintenance, and quality assurance. Depositions were reviewed as were a large number of documents including such items as I&E inspection reports, TMI procedures, licensee event reports (LERs), hardware history, meeting minutes, etc. All persons contacted were very helpful and open and appeared to have a genuine desire to determine and correct the cause of the accident.

As the review proceeded, it also became evident that detailed study of certain hardware involved in the accident, namely the condensate polisher, the iodine filter process, the pilot-operated relief valve (PORV), and emergency feedwater valves was needed. The results of these detailed investigations are reported in separate staff reports, but their results provide significant support to the findings in this paper.

III. ANALYSIS

A. REQUIREMENTS AND REGULATIONS

1. General Quality Requirements

The applicant for a construction permit to build a nuclear power plant or for an operating license to operate one is required to submit a Safety Analysis Report (SAR) to the NRC based on the requirements of 10 CFR 50 (reference 1). The SARs cover all aspects of the proposed plant from the point of view of safeguarding the public health and safety, ranging from detailed design considerations to operating considerations. The SAR is reviewed by the various branches of the Office of Nuclear Reactor Regulation (NRR) for areas under their cognizance, coordinated by a project manager who serves as the main point of contact with the applicant during the licensing process.

Today the review of the SAR by NRC is conducted in accordance with a Standard Review Plan (SRP) (reference 2), a 1,100-page document which identifies acceptance criteria for each area reviewed. At the time of the TMI-2 SAR review, the SRP was in the process of being implemented. NRC personnel contacted felt that SRP requirements were generally followed for TMI-2, but any differences would generally result in a less stringent review. Roger Mattson indicated (reference 12) and Commission staff review has confirmed, that TMI-2 was "grandfathered" for a number of current SRP and licensing requirements; however, for the purpose of evaluation of the NRC/TMI overview process, this staff review references the SRP since it is documented and any weaknesses identified would apply to both TMI-2 and the current NRC review practices.

When all the acceptance criteria of the SRP are met, a Safety Evaluation Report (SER) is prepared by NRR, which upon verification that the programs are being implemented, is submitted to the Advisory Committee on Reactor Safeguards (ACRS) for their independent review -- first by a subcommittee, and then by the full ACRS. The findings of the ACRS are reported to the chairman of the NRC. After hearings by the Atomic Safety and Licensing Board (ASLB) result in a favorable finding, the applicant is issued either a construction permit or an operating license.

In 1970, the Atomic Energy Commission -- predecessor to NRC -- added 10 CFR 50, Appendix B (reference 3), to the already existing regulations. Appendix B is a general requirement for a quality assurance program for the design, construction, and operation of nuclear power plants. This regulation was intended to clarify vague references to quality assurance requirements elsewhere in the regulations. It specifies 18 criteria dealing with organization, the utility quality assurance program plan, and aspects of design control, document control, test control, corrective action, and the like.

Chapter 17 of the SAR is now devoted to the applicant's quality assurance program, which is reviewed by the NRC Quality Assurance Branch (QAB) in accordance with the SRP acceptance criteria for that chapter.

The 18 criteria of 10 CFR 50, Appendix B, are very broad and general, occupying about two pages of 10 CFR 50. To assist applicants in preparing their programs, a large body of supplemental guidance has been prepared. Regulatory guides have been prepared which identify an acceptable way of complying with the provisions of Appendix B. (Although other ways may be acceptable, their departure from ways already reviewed by NRC requires additional review time, a costly commodity. Consequently, it is usual to find the programs in the regulatory guides accepted by the applicants.)

The regulatory guides normally endorse industry standards; in the quality assurance area, they endorse consensus standards that are developed under the auspices of the American National Standards Institute (ANSI) and that are prepared with participation by industry, technical societies, and the NRC. For design and construction of nuclear plants, the ANSI standard N45.2 (reference 4) is endorsed, together with a number of associated "daughter" standards which address specific quality assurance practices. For operational quality assurance programs, ANSI N18.7 (reference 5) is endorsed.

As discussed in section III-F of this report, the basic quality assurance requirements are very similar to those used in quality assurance programs for other safety-critical industries and could be adequate for the quality program for nuclear power plants, if properly applied. As discussed and illustrated throughout this paper, the weakness in the regulations is primarily associated with their application and enforcement. As written, the regulations do not ensure a strong, independent quality assurance program or organization, primarily due to their limited scope and their general philosophy that allows other than the quality organization to conduct independent assessment of activities normally reviewed by quality programs.

Finding

- Quality assurance requirements, as stated in 10 CFR 50, Appendix B, appear adequate for those systems to which they apply.

2. <u>Quality Assurance for Plant Systems and Operations</u>

a. Quality Requirements for Plant Systems

10 CFR, Appendix B, applies to "structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public" (reference 3).

According to W. M. Morrison, assistant director for General Engineering Standards at NRC (reference 35) and one of the authors of 10 CFR 50, Appendix B, this statement was intended to paraphrase statements in 10 CFR 50, Appendix A (reference 6), which speak of "quality standards commensurate with the importance of the safety functions to be performed." Morrison explained that Appendix B was intended to impose quality assurance on all portions of the plant that could affect safety, but allow a graded approach in which the degree of control was commensurate with the item's importance to safety.

In application, however, both NRC and the industry have interpreted 10 CFR 50, Appendix B, as applying only to structures, systems, and components identified as "safety-related."

The determination of whether or not an item is safety-related is based partly on the application of the "single failure criterion," to each of 39 design-basis accidents (Standard Review Plan for chapter 15 of the Safety Analysis Reports). The determination also considers whether the item is designed for seismic loads (reference 7) or is part of the pressure boundary system.

A significant flaw in the NRC guidance regarding the determination of what is safety-related is the limitation of safety-related to the function of equipment installed primarily for safety.

As a consequence, function of equipment associated with normal operations, such as the pilot-operated relief valve (PORV), the condensate polishers, or the thermocouples in the reactor core (reference 8) are not considered to be safety-related, although the role of such equipment in the TMI-2 accident has proven to be significant. Also, by restricting safety-related to protective devices, equipment like the radiation monitoring equipment does not qualify as safety-related.

This sharp differentiation between safety-related and non-safety-related means that only the safety-related items are automatically covered by the quality assurance requirements of 10 CFR 50, Appendix B. Quality assurance requirements for the remainder of the plant depend on individually specified, and usually incomplete requirements scattered throughout the regulations. A corollary effect is that the quality of many items significant to the control of accidents now falls outside NRC control.

A list of safety-related systems is included in the Safety Analysis Reports and is thereby subject to NRC review. This generally amounts to "several dozen items" (reference 9). Detailed lists of individual safety-related valves, pipes, etc., contained in these systems were prepared for TMI-2 by the architect engineer (AE) and are available at the plants, but, as found by Commission staff, these lists were not contained in the Final Safety Analysis Report (FSAR).

Therefore, the detailed lists of what equipment is safety-related may not be examined indepth by NRC, and the Commission staff review indicates they may be generally unfamiliar to operating plant personnel. For example, the supervisor of TMI's quality control activity was unsure whether the PORV was safety-related or not, yet this would determine whether the quality organization should be involved in maintenance activities associated with it.

Finding

- Quality assurance applies only to a narrow portion of the plant defined as safety related or safety grade. Many items vital to the safe and reliable operation of the plant are not covered by the quality assurance program because of this definition.

b. Quality Requirements for Operations

Regulatory Guide 1.33, Quality Assurance Program Requirements for Operation (reference 10), provides guidance to licensees regarding the development of a quality assurance program for plant operations. It endorses ANSI N18.7/1976 (reference 5) as describing a quality assurance program for plant operations that is, with minor modifications identified in the guide, acceptable to the NRC as a way of complying with 10 CFR 50, Appendix B. (For design and construction of nuclear power plants, regulatory guides endorse ANSI N45.2 and its "daughter" standards.)

A major finding pointing up the need for assurance of operations is contained in the forward to ANSI N18.7 where it is stated that because of the dynamic nature of operations, operating deficiencies can be much more immediate in their effect than deficiencies associated with design or construction. But in spite of the recognized critical nature of operations the ANSI N18.7/1976 entrusts the responsibility for verifying conformance to requirements of the technical specifications and other drawings, instructions, and procedures for operating plants, to second-line supervisors or other qualified personnel not assigned firstline supervisory responsibility for conduct of the work. Independent quality assurance personnel would only be involved in nonroutine maintenance and modification activities where it is not considered necessary.

This is a clear, though perhaps unintentional, indictment of the quality assurance programs in operating plants. It has not only removed the review of operations and surveillance activity from the TMI-2 quality organization, but as described in subsequent sections, has also removed from the quality organization the review of operating and surveillance procedures used in such activity.

Section 4.3 of ANSI N18.7 does provide that an independent review shall be conducted of "activities occurring during the operational phase -- on a periodic basis". The requirements for the organization to conduct this review are defined as are the subjects requiring independent review. This is an off-site group including competence in the various technical disciplines and quality assurance. The group is not required to review day-to-day activities and the on-site personnel make basic decisions as to whether changes do involve technical specification or unreviewed safety questions which should be then reviewed by the off-site group. In the case of TMI, the Generation Review Committee (GRC) provides the independent review. On-site review is conducted by the Plant Operating Review Committee (PORC) -- made up of personnel from the operating organization -- and does not include quality assurance of other independent assessments.

While it is certainly important to have knowledgeable people conducting on-site reviews, and also to have reviews conducted by personnel not directly responsible for the tasks, it is perhaps even more important that the reviewers have independent motives and mental sets -- continuing operation of the plant is probably not the primary motive the independent reviewer should have. Rather, the reviewer must assure the plant will be operated safely and in rigorous accordance with applicable regulations, technical specifications, and licensing requirements.

While historically independent review of a given task has been considered inspection or quality functions, other forms of independent assessments are used. Based on staff experience, it is known that industrial safety organizations which overview the day-to-day operation of a manufacturing plant are standard practice. The joint Atomic Energy Commission/National Aeronautic and Space Administration (AEC/NASA) nuclear rocket program, operating under AEC rules and regulations (not licensed), used an independent safety office in both government and contractor organizations to provide an independent assessment of facility and research test operations. Persons in this role were highly qualified nuclear or cryogenic engineers who had safety training and experience and had prime responsibility for looking for potential weaknesses, hazards, and problems in the design and operation.

Subsequent sections will indeed show that the attention given by operating personnel, the Plant Operations Review Committee, and higher level review groups to operating conditions and procedures at TMI-2 was inadequate and that some form of on-site independent review group is required.

Finding

- There is no requirement for independent, on-site quality or safety assessment of operations. Surveillance testing by the utility is audited infrequently.

Regulations allow review to be done by in-line supervision and other personnel directly responsible for operations.

3. Reliability and Safety

a. Responsibilities of Safety and Reliability

The function of reliability and safety in complex, safety-critical programs and systems is to perform continuing analyses of the total system in use, to determine hardware and human failure modes, their influence on system behavior, and, to the extent possible, the likelihood of their occurrence. In these programs (reference 110, 111, 120), results of such analyses are used to:

- identify needed design changes to prevent or accommodate the more likely or more significant failure modes, both those found by analysis and those found by experience;

- identify, through examinations of the role of human operator and maintenance actions, requirements being placed on the human which unduly tax his capabilities and which are significant to system behavior;

- address interactions between the above two -- identify needs to redesign to alleviate problems arising because of operator or maintenance limitations, and identify operator or maintenance requirements resulting from hardware or system limitations;

- establish data requirements for tests and operations which will identify the occurrences of failure modes;

- identify the significance of hardware to reliability and safety, so that appropriate quality assurance programs can be applied; and

- identify the significance of human actions (operator, maintenance, etc.) so that training and procedure development can be properly performed.

As part of the continuing analyses, it is necessary that the reliability/safety organizations, in conjunction with engineering, collect and analyze applicable test and operating data to maintain current, refined assessments (qualitative or quantitative) of the failure modes and the likelihoods of occurrence. This requires continuing interaction with data collection activities, as experience will result in clearer, more precise descriptions of the failure modes, and will pinpoint weakness as in design, procurement, operation, and manufacturing practices which contribute to the failure.

It is also necessary that the reliability/safety function be independent of design-related activities to preserve its objectivity, that it have top management support and clear access to top management to resolve disputes, since all the actions which affect safety are taken by other organizations (design, training, quality assurance, etc.), some of which reflect opposing interests. It is also necessary for top management to understand the assurance roles of quality, reliability, and safety in order to make effective decisions based on their findings.

The following discussion reviews the NRC's regulations as they now exist and shows that an overall safety/reliability program is not a requirement of the regulations.

Further sections of this report illustrate that the lack of such a requirement resulted in procedures and practices that were a significant part of the cause of the TMI-2 accident.

b. Safety and Reliability Analysis Requirements

Unlike government-contractor relationships, the NRC's control of nuclear power plants is only partial. NRC is not empowered to address the reliability of nuclear power plants from the point of view of power availability. However, nuclear power plant safety depends in large measure upon active protective systems such as the emergency core cooling system (ECCS), so that the reliability of these features unquestionably falls within NRC's scope of regulation.

Additionally, the investigation of the TMI-2 accident clearly showed that the reliability of nonsafety grade plant hardware associated with normal operations -- and which NRC has previously excluded from regulatory control -- can have a significant influence on plant safety. It is being recognized by both the utility industry and the NRC (reference 25) that whether one is interested in plant safety or plant availability, the behavior of all the plant hardware and operations must be considered.

It is for these reasons that reference is made herein to reliability/safety analysis; a combined activity in which techniques from both disciplines -- which are merely variants of the same basic approach -- are brought to bear on plant safety, addressing the entire plant.

The above does not imply that NRC does not address safety; that is all they <u>do</u> address. Engineering analysis of plant safety features is NRC's strongest point, and this is amply reflected in the regulations. Instead, what is addressed here are the technical activities which analyze the safety of the operating plant, equipment and operators as an entity, and which would provide for more complete treatment of hardware and human failures at both system and subsystem levels.

Recommendations have been made persistently to NRC to adopt formal reliability/safety practices (see below), but NRC has not seen fit to recognize such practices even to the extent they did quality assurance practices when they extablished 10 CFR 50, Appendix B. Hence, reliability/safety analysis requirements are not definitive, but are distributed throughout the regulations in much the same way quality requirements were prior to the addition of 10 CFR 50, Appendix B. Similarly, the reliability/safety issues are related to subsystem design, and where reliability or safety techniques of failure mode and effects analysis, or hazard analysis, are applied, it is done on specific components or subsystems. As a result, each licensing branch reviews those issues within its specialty area (reference 11) and in accordance with acceptance criteria in the Standard Review Plan (SRP) (reference 12).

Where safety and reliability analysis is invoked, design emphasis is on the "single failure criterion" as a basic reliability/safety principle. The single failure criterion states that safety-related fluid and electrical systems are to be designed so that "neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive components (assuming active components function properly), results in a loss of the capability of the system to perform its safety function" (reference 6).

Application of the single failure criterion is on a system-by-system basis by successive paragraphs of 10 CFR 50, Appendix A, which addresses the safety-related structures, systems, and components of the plant.

The single failure criterion is an inferior rule of design by comparison with more extensive reliability/safety techniques available. For example, a 1974 study by NASA's Johnson Space Center for the Electric Power Research Institute (EPRI) of the "Application of NASA Safety, Reliability and Quality Assurance Techniques to the Nuclear Power Industry" (reference 13), cited a "lack of identification and analysis of hardware failure modes" and recommended the application of failure mode and effects analysis (FMEA) during the design of nuclear power plants. This recommendation has been fulfilled for some subsystems, but not generally applied and its results utilized in a systematic way.

In a letter from the Institute of Electrical and Electronic Engineers (IEEE) to Saul Levine of NRC's Office of Nuclear Regulatory Research (RES), dated April 1976 (reference 14), the IEEE stated that "ritualistic applications of single failure criteria" do not serve the public safety purpose. Also, the report of a Sandia Laboratories study of the NRC quality assurance program, published in 1977, indicated that the "redundancy imposed by the single failure criteria may be inappropriate" (reference 15).

Finding

- o Reliability/safety analysis requirements are applied to specific safety-related hardware as specified in 10 CFR 50, Appendix A, utilizing questionable "single failure" criterion.

c. The Role of "Safety-Related" in Plant Reliability and Safety

Another ramification of the single failure criterion is its use, as discussed previously in conjunction with postulated accident sequences, to establish which items are to be designated as "safety-related."

The designation of safety-related brings the force of NRC regulations to bear; the design of safety-related items is examined in excruciating detail, the provisions of 10 CFR 50, Appendix B, apply, I&E inspectors continually check on these items, and operating problems involving such items must be reported in licensee event reports.

On the other hand, items which escape the designation as safety-related are accorded casual treatment, if any. Design and failure modes reviews are conducted only to the extent necessary to assure they are not needed for plant safety and will not prevent the safety-related systems from working (reference 25). While they may be required to meet standard design codes they need not be testable in the system; they do not require redundancy, they are not subject to quality requirements, and are not ordinarily subject to I&E inspection (references 16, 12, and 17).

To illustrate the significance of this point, on March 29, 1978, the PORV at TMI-2 failed, causing a shutdown of the plant. The regional inspector requested review of the design approach (valve failing open on loss of power). The reply he received was that the design was determined to be acceptable in view of the TMI-2 Final Safety Analysis Report. No other rationale was given. During his deposition, Seyfit was questioned on this reply:

> QUESTION: Do you recall the reason that it was concluded that PORV failing in the open position was an acceptable design feature of TMI-2?
>
> SEYFRIT ANSWER: The major one was that the high pressure safety injection system was sized to be able to provide water to the reactor at a rate greater than could be lost through the open PORV. So that there was indeed a back-up system in the event of a failure. And based on the single failure criterion which has been used traditionally, that would make it an acceptable design.
>
> QUESTION: In other words, the assumption would be that single failure of the PORV would not result in core uncovery because no failure with respect to ECCS was built into the analysis?
>
> SEYFRIT ANSWER: That is correct. (reference 18)

The PORV was not a safety-related item (reference 19) because it had a block valve to isolate it from the primary system. The block valve was not a safety-related item because it had a PORV in series with it (reference 12).

A more detailed discussion of the ramifications of the application of "safety-related" to the nuclear power plant program is discussed in section III-B of this report. Our review of requirements indicates, however, that the application of the safety-related philosophy has severely limited requirements for, and subsequent benefits of, a judicious use of safety and reliability techniques.

Finding

- Safety and reliability requirements and analyses are not required to be applied to many plant systems which may be "vital" to the safe operation of the plant, but are not labeled "safety related."

d. Safety and Reliability Organizations

In other safety critical programs, the requirement to conduct reliability and safety analysis resulted in the development of expertise in these areas. The continued technical competency of these organizations is assured simply by their need to conduct continuing reliability/safety analyses, based on the need to conduct design reviews of new hardware while obtaining feedback from operational test data on previous designs. These continuing analyses constitute significant independent reviews of designs and their operational ramifications.

Since the reliability/safety requirements that do exist in the regulations are primarily related to component's and subsystem's reaction to the design base accidents and are under the control of engineering specialty groups at NRC and in industry, there has been no apparent need felt in NRC or industry to establish such reliability and safety organizations. Consequently, within the industry, designs may be "independently" reviewed by other designers (and reviewed by design experts at NRC); operational conditions and procedures may be "independently" reviewed by other operating plant personnel (reference 20). This virtually assures that no independent review is established to catch technical mistakes, not to mention mistakes resulting from job-related pressures or orientations -- keep the plant operating, pride in the design, etc.

Historically, as the discipline of safety and reliability developed in the aerospace and other safety/reliability-conscious industries, the organizations evolved in conjunction with quality assurance at a level in the organizational structure generally equal to that of engineering, so that reliability and safety concerns surface when senior management is involved in their resolution. This has not happened in the nuclear utility industry.

Finding

- Lack of requirements by NRC in the safety and reliability disciplines has resulted in little motivation for strong safety and reliability engineering capability in NRC and the utility industry.

e. Human Factors

An explicit part of the reliability/safety modeling, failure mode effects analysis (FMEA), and hazard analysis is the treatment of human factors associated with operations and maintenance. Normal practice identifies human failures as explicit events in fault tree/event tree models along with hardware failure events. Human factors specialists are employed to identify errors, likely situations in operational and maintenance activities, and to interact with designers and planners to change these situations so as to avoid over-taxing the human or modifying his/her influence on the system.

There have been at least two studies performed during the last several years regarding human factors considerations in reactor control rooms: one by the Lockheed Corporation for EPRI and one by Aerospace Corporation for the NRC but results of these studies have not found their way into licensing operations. Recommendations were provided to the NRC Office of Standards Development (SD) for incorporation into regulatory guides, but they have apparently not been followed by upgrading the acceptance criteria used in licensing reviews.

QUESTION: Is there any office within the NRC that looks at the man-machine interface...?

MATTSON ANSWER: No.

QUESTION: Why has it become the situation then that certain types of accidents are simply not anticipated or designed against today?

MATTSON ANSWER: ...[P]eople believed evidently in providing safety systems, well engineered, well designed, well analyzed safety systems, and the fault was they believed so much in the infallibility of the safety systems, they forgot about the people who could stand by and defeat them if they didn't have the right training. (reference 12)
. ..[W]e in the Systems Branch did not specifically look at the operator actions. It may now in hindsight be a weakness or a specialization by which the Branch did its work that it was not able to really put the operator in the systems review proce (reference 11)

The failure to take the human factor into account during plant licensing contributed directly to the TMI-2 accident. For example, it has been suggested that the TMI-2 operator could have discovered that the PORV was stuck open, despite failure of the PORV position indicator to indicate valve position, by observing the quench tank temperature and pressure indication (reference 134). However, the quench tank indicators at TMI-2 are located on the back of the control room panel, and, therefore, are not readily available to the operator (reference 17). Locating such indicators on the back of the control panel is not a violation of any NRC design requirement (reference 17). In fact, NRC has no specific requirements at all regarding control room layout (reference 22), and NRC Division of Systems Safety (DSS) does not review control room design (reference 12).

Finding

- o Present NRC design, safety, and reliability requirements do not generally address human factors and the man-machine interface.

B. THE OPERATIONAL QUALITY ASSURANCE PROGRAM

The responsibility of the NRC has been defined in several ways. One definition used by the NRC is as follows:

> The Commission's primary concern for safety in civilian nuclear activities involves two major considerations: the risks imposed by serious nuclear accidents on the one hand, and by exposure to routine releases of radioactivity on the other. . . The Commission's safety goal, implemented from guidance from national radiation protection standards, is to see that its licensees and applicants for licenses take the actions considered necessary to assure that there are no undue risks to the public and workers from both normal activities and potential accidents.

Another definition, from the document "U.S. Nuclear Regulatory Commission Functional Organization Charts" (reference 24), is as follows:

> The Commission is responsible for licensing and regulating nuclear facilities and material and for conducting research in support of the licensing and regulatory process, . . . These responsibilities include protecting public health and safeguarding materials and plants in the interest of national security; and assuring conformity with antitrust laws. Agency functions are performed through: standards-setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits and licenses; inspection, investigation and enforcement; evaluation of operating experience, and confirmatory research. The Commission is composed of five members, appointed by the President and confirmed by the Senate, one of whom is designated by the President as Chairman. The Chairman is the principal executive officer and the official spokesman of the Commission.

In these definitions, emphasis is on safety; safety of the public and safety of the workers associated with the activity. It is the intent of this portion of the analysis to evaluate the NRC and utility organizations and their performance that resulted from the NRC regulations discussed previously to determine if this safety responsibility is being adequately accomplished by the NRC and utility staffs. This evaluation will be accomplished by examining the project management, engineering, safety, reliability, and quality assurance responsibilities assigned to each major line organizational element of the NRC and utility applicable to the overview process and evaluating the performance of each, both in meeting basic NRC requirements and in meeting safety needs not delineated by the requirement documents. The evaluation will weigh what is being done and how it relates across the organizations, more than where in the organization it is assigned.

1. <u>NRC Organization and Responsibilities</u>

A chart showing the overall NRC organization is shown in Figure 1. The organizations, which are of interest to this evaluation and which, from the assigned responsibilities, appear to be closest to the reviews of utility qualifications leading up to the granting of the operating license and also close to the monitoring of the operating plants, are located in the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement (I&E), Figures 2-7. The Office of Standards Development, Figure 8, provides and controls the basic standards used by the groups involved in the overview process and therefore need a feedback mechanism to determine the effectiveness of their standards. They are not, however, participating in the direct overview process.

The Office of Nuclear Material Safety and Safeguards (NMSS) is involved in nuclear fuel and security overview, but these functions are out of the scope of this evaluation.

The responsibilities assigned to the organizations of interest are shown in Figures 2 through 8, as they are presented in reference 24. In addition, the responsibilities of the Advisory Committee on Reactor

Safeguards (ACRS) is shown on Figure 9 as it has a direct function in the granting of the license, but is not part of the day-to-day overview process. The Office of Management and Program Analysis (MPA) is shown in Figure 10. Relative to this evaluation, MPA is primarily involved **in** the processing of licensee event reports (LER). In general, these responsibilities seem to be carried out as follows:

- The Advisory Committee on Reactor Safeguards, Figure 9, reviews license applications, generally studies and reviews subjects brought to their attention, and provides recommendations as appropriate, from their position as an advisory function (Reference 31).

- Divisional elements within the Office of Nuclear Reactor Regulation, Figures 2-5, are the line organizations which review and analyze applications for operating licenses, grant these licenses when requirements are satisfied, and review the operating performance of the licensed utilities.

- Within the Division of Project Management (DPM) is located the Quality Assurance Branch (QAB) which "reviews reactor license applications to assure compliance with quality assurance criteria during plant design, construction, and operation and evaluates technical and administrative competence of reactor operating organization for protection of public health and safety," Figure 3.

- The Office of Inspection and Enforcement, Figures 6 and 7, is that office of NRC that maintains "physical" contact with each utility by having its personnel on utility sites to provide information through inspection that the utility is complying with regulation and meeting the issued technical specification.

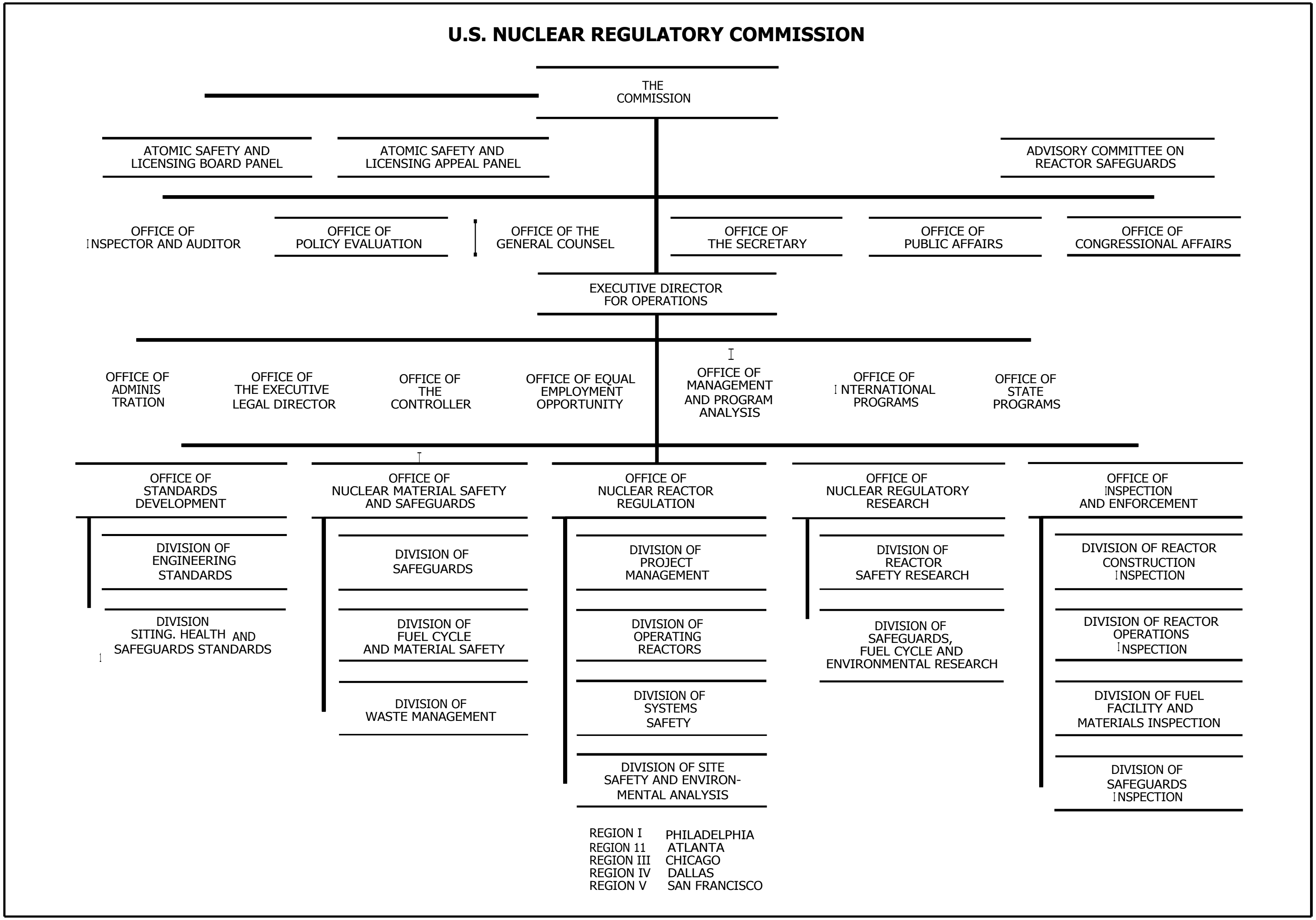
a. Traditional Responsibilities Not Reflected in NRC Functions

In examining the NRC organization it has been noted that certain responsibilities traditionally found in organizations that manage and/or monitor high risk activities appear to be missing. These responsibilities are:

1. The assignment to provide and enforce rigorous problem reporting and failure analysis and corrective action, and one that systematically considers changes to training and procedures as well as to hardware.

The functional assignments of critical nuclear power plant overview organizations such as DPM, including the various reactor branches, Quality Assurance Branch, and Operator Licensing Branch, all branches within DOR and DSS neglect this function. Even the I&E chart neglects this function even though I&E is responsible for general overview of LERs. The Licensee Operations Evaluations Branch within MPA reviews LER, but primarily to enter them into the NRC computer system (reference 96).

FIGURE



Source: NUREG-0325, "U.S. **Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision** 1, Jan. 1, 1979.

FIGURE 2

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Performs licensing functions associated with construction and operation of nuclear reactors and with receipt, possession, ownership, and use of special nuclear and byproduct material used at reactor facilities; reviews applications and issues licenses for reactor facilities required to be licensed under the Atomic Energy Act of 1954, as amended, and evaluates health, safety, and environmental aspects of facilities and sites; develops and administers regulations; licenses reactor operators; provides assistance In matters involving reactors or critical facilities exempt from licensing; analyzes reactor design concepts; evaluates methods of transporting nuclear materials and radioactive wastes on reactor sites; and monitors and tests operating reactors, recommending upgrading of facilities and modification of regulations, as appropriate.

Director — Harold R. Denton
Deputy Director — Edson G. Case

PROGRAM SUPPORT STAFF

Provides technical assistance and support to the Director in evaluating proposed office activities or positions and in the administration of highly technical and diversified licensing projects. Plans, coordinates, directs and executes such administrative affairs as contractual arrangements and agreements; budget formulation; fiscal management surveys and directives; personnel administration, development and training; and general office services. Develops, plans and implements programs to assess and improve the effectiveness and efficiency of the licensing process; coordinates NRR-related confirmatory research programs and assesses their effectiveness; develops, plans, coordinates and Implements generic technical assessments of existing and proposed office safety and environmental criteria and requirements.

Director — 0 F Bunch

ANTITRUST AND INDEMNITY GROUP

Establishes and administers antitrust reviews of license applications for nuclear facilities, including compliance with license conditions pertaining to antitrust matters; indemnification of licensees against public liability claims arising out of nuclear incidents; and execution of indemnification agreements with licensees.

Chief — Jerome 0 Saltzman

EcononucsSection

Reviews, investigates and evaluates existing and potential anticoepetitive activities by applicants; prepares testimony relating principles of competition and monopoly to antitrust cases; assists in policy formulation, and prepares studies and analyses on economic issues of antitrust cases.

Chief — Maurice Messier

Power Supply Analysis Section

Investigates and prepares related testimony concerning power supply and intersystem cpordination activities by applicants; assesses impact of potential license conditions; provides technice) assistance on engineering and power supply aspects of antitrust cases; carries out compliance activities.

Chief — Argil L Toalston

DIVISION OF PROJECT MANAGEMENT

law purge 34)

DIVISION OF OPERATING REACTORS

Isee page 351

DIVISION OF SYSTEMSSAFETY

(we page 361

DIVISION OF SITE SAFETY AND ENVIRONMENTAL ANALYSIS

(mm pope 37)

Source: NUREG-0325, "U.S. Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.

FIGURE 3

U.S. NUCLEAR REGULATORY COMMISSION

DIVISION OF PROJECT MANAGEMENT

Carries out project management functions for reactor safety reviews through operating license stage; coordinates and schedules adequate and consistent technical reviews for each application.

Director — Roger S. Boyd
Deputy Director — Denwood Ross

ASSISTANT DIRECTOR FOR LIGHT WATER REACTORS

Plans, directs and supervises programs and activities of Light Water Reactors Branches, including evaluation of the operational safety and design modification of DOE-owned and DOD-owned reactors exempt from licensing,

Assistant Director — Domenlc B Vassallo

ASSISTANT DIRECTOR FOR STANDARDIZATION AND ADVANCED REACTORS

Plans and directs NRC's Standardization program and project management functions for standard plant design reviews and for review of applications for construction permits, operating licenses and amendments, and licensability reviews for fast *breeder,* gas-cooled and other advanced power reactors, plans and directs reviews of military and non-military nuclear propulsion and power systems and nuclear-related facilities; plans and directs safety reviews of certain aspects of fast breeder, gas-cooled and other advanced reactor concepts. Develops and implements standardization policy and procedures.

Assistant Director — William P Gainer ill

ASSISTANT DIRECTOR FOR QUALITY ASSURANCE AND OPERATIONS

Plans. directs and supervises programs and activities of the Duality Assurance, tmergency Planning, and Operator Licensing Branches; reviews financial qualifications of applicants and licensees; prepares hearing testimony on financial qualifications; specifies license conditions to assure utility commitments for plant construction and operation.

Assistant Director — Donald J. S orholt

Light Water Reactors Branch Y1

Reviews. analyzes and evaluates application for construction permits and operating licenses for assigned reactors and for certain reactors exempt from licensing; evaluates technical specIftcattons for such reactors. performs safety review of assigned operating reactors. including review of amendments to licenses changing technical specifications.

Chief — Jnhn F. Stolz

Light Water Reactors Branch *2

Reviews, analyzes and evaluates applications for construction permits and operating licenses for assigned reactors and for certain reactors exempt from licensing; evaluates technical specifications for such reactors; performs safety review of assigned operating reactors, Including review of amendments to licenses changing technical specifications.

Chief — *Robert* L. Baer

Light Water Reactors Branch *3

Renews, analyzes and evaluates applecatiuns for construction permits and operating licenses for assigned reactors and for certain reactors exempt t'om licensIv,g, evaluates technical specifications for such reactors performs safety review of assigned operating reactors, includirg review of amendments to licenses changing technical specifications,

Chief — *Olin* D Parr

Light Water Reactors Branch ma

Reviews, analyzes and evaluates applirations for construction permits and operating licenses for assigned reactors and for certain reactors exempt from licensing; evaluates technical specifications for such reactors; performs *safety* review of assigned operating reactors. Including review of amendments to licenses changing technical specifications.

Chief — Steven A Varga

Sundardlzation Branch

Coordinates Standardization program including development and implementation of policies and procedures and project management functions associated with safety reviews for approval of designs of standard plants and major portions thereof, plant systems and components; provides project management functions for reviewing design, construction and operation of licensed nuclear power reactors and for other military and non-military government nuclear systems and facilities.

Chief — Clemens J. Neltemes

Advanced Reactor Branch

Evaluates design and performance of fast breeder, gas-cooled and other advanced reactor concepts; manages safety reviews of applications to construct and operate such reactors; performs licensabtilty evaluations of advanced reactor concepts; *reviews* and evaluates advanced reactors exempt from licensing, including technical specifications for such reactors,

Chief — heals P Specs

Quality Assurance Branch

Reviews reactor license applications to assure compliance with quality assurance criteria during plant design. construction and operation; evaluates technical and administrative competence of reactor operating organization for protection of public health and safety.

Chief — Walter P. Naass

Operator Licerminp *Branch*

Examines and licenses candidates *for* reactor operator and senior reactor operator licenses; develops qualifications requirements, testing techniques, and standards for evaluation of candidates and Conducts safety evaluations for design and operation of reactor projects,

Chief — Paul F Collins

Source: NUREG-0325, "U.S. **Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.**

FIGURE 4

U.S. NUCLEAR REGULATORY COMMISSION

DI VISION OF OPERATING REACTORS

Reviews the design and operational changes in operating reactors; analyzes and responds to operating experiences as they develop; and assures that current experience is factored into new licensing actions.

Director Victor Stello
Deputy Director Dnrell G. Eiminhut

ASSISTANT DIRECTOR FOR SYSTEMS AND PROJECTS

ans and . rects tec nice sa e y review a ' eva ua on o licensed operating power reactors, licensed test, research and critical facilities, and review of proposed amendments to operating licenses; evaluation of operational safety and design modifications of DOE- and 000-owned reactors exempt from licensing; technical safety review and evaluation of reactor systems and plant systems; technical review of applications and issuance of construction permits and operating licenses for no ,, r reactors.

Assistant Director Vrant

Systematc v uđ̃ — Progr Branch

Performs multiple discipline technical evaluations of older operating power facilities to reconfirm and document their safety and to determine extent to which they meet current licensing requirements. Establishes technical guidelines which may require plant or procedural modifications for selected facilities for particular safety considerations.

hief Donald K Dav,

Reactor Safety Branch

Performs engineering safety reviews, analyses and evaluations of reactor coolant systems, engineered safety features, an. accidents and transients for all reactor facilities licensed for operation; evaluates applications for construction permits and operating licenses for nonpower reactors a evaluates operational and design modifications of DOE- an. DOD-owned operating facilities exempt from licensing as requested.

Chief Paul S Chec

Plant Systems Branch

'er orms ec n ca sa e y review an- eva ua on o ..i ica-t ions to plant systems for all reactor facilities licensed for operation, of all applications to construct and operate no npowr reactors, and, as requested, of operational and design modifications of DDE- and D00-owned reactor facilities exempt from licensing; performs technical safety review and evaluation of contairrent systems, power conversion systems, auxiliary systems, electrical systems, control systems and all other safety-related instrumentation for above facilities

Chief Gus C Lainas

Operating Reactors Branch 1

Performs overall project management for a signed licensed power operating reactors including review of technical and procedural aspects involving proposed amendments to operating licenses.

hief Albert Schwencer

ASSISTANT DIRECTOR FOR REACTOR SAFEGUARDS

Plans, directs and administers NRR reactor safeguards program; and provides management direction and supervision to the Reactor Safeguards Licensing and Reactor Safeguards Development Branches.

Assistant Director James R. Miller

Reactor Safeguards Licensing Branch

Reviews safeguard-related sections of PSAR's and FSAR's, applicants' physical security plans and contingency plans, and amendments to these plans for operating reactors; participates in operating reactors re-reviews; and directs technical assistance programs.

Chief Robert A Clar

Reactor Safeguards Development Branch

Develops reactor safeguards policy recommendations; develops and analyzes new regulatory requirements involving safeguards; coordinates research and development programs; develops evaluation criteria for use in licensing function; identifies and initially develops required standards; and prepares and maintains NRR/IE reactor safeguards response plans.

Chief Frank G. Pagano

Opermniog Reactors Branch #2

Performs overall coordination and project management, including technical and procedural aspects, involving proposed amendments for those operating power reactors which are under re-review as part of MRC's program to systematically evaluate margins of safety of design and operation with respect to current licensing requirements.

Chief Dennis L. Ziemann

Source: NUREG-0325, "U.S. Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.

FIGURE 5

U.S. NUCLEAR REGULATORY COMMISSION

DIVISION OF SYSTEMS SAFETY

Carries out detailed safety reviews of reactor applications through the operating license stage.

Director — Roger J. Mattson
Deputy Director — Frank Schroeder

ASSISTANT DIRECTOR FOR ENGINEERING

Plans, directs and supervises the programs and activities of the Branches listed below.

Assistant Director — Ja s P. night

Mechanical EngrneeringBraadh

Reviews and evaluates seismic and pipe whip design and mechanical design of reactor vessels, reactor core supports, fuel components, coolant pumps, steam generators, coolant piping, pressurizers, component supports and other safety-related mechanical components.

Chief — Robert J. Bosnak

Mnanrais Engineering Branch

Evaluates materials of pressure-retaining components of fluid systems important to safety; performs general technical review, analysis and evaluation of materials, fabrication, inspection and testing of reactor components and systems.

Chief — Stefan S. Pawlicki

Structural Engineering Branch

Evaluates missile design, design of concrete and steel containments, and design of other safety-related plant structures; performs technical review, analysis and evaluation of design, construction and operation of nuclear power reactor structures.

Chief — Franz P. Schauer

ASSISTANT DIRECTOR FOR REACTOR SAFETY

Plans, directs and supervises the programs and activities of the Branches listed below.

Assistant Director — Robert L.Tedesco

Reactor Systems Branch

Reviews and evaluates design and performance of reactor thermal-hydraulic systems, reactor coolant systems, and associated auxiliary systems, and emergency core cooling systems.

Chief — Thomas M. Novak

Core Performance Branch

Reviews, develops and executes calculational methods in the physics, thermal and hydraulic, and reactor fuel aspects of nuclear reactor design.

Chief — Karl Kniel

Analysis Brach

Reviews, evaluates and analyzes calculational methods used by applicants for licensing of nuclear power plants in the nuclear, thermal, and hydraulic areas of reactor and engineered safety features design; develops, in conjunction with consultants, independent calculational methods, including complex computer codes, for analyzing nuclear, thermal, and hydraulic performance during steady-state, transient and accident conditions.

Chief — Zoltan R. Rosztocty

Conniranant Syrtans Branch

Reviews reactor license applications and related documents to evaluate containment systems and associated sub-systees, including heat removal, heating and ventilation, isolation equipment and controls, and combustible gas control systems.

Chief — Walter R. Butler

ASSISTANT DIRECTOR FOR PLANT SYSTEMS

Plans, directs and supervises the programs and activities of the Branches listed below.

Assistant Director — Stephen H. Hanauer

Auxiliary Systerns Branch

Reviews reactor license applications and related documents to evaluate the design, fabrication and operation of auxiliary systems and fire protection programs.

Chief — Victor Benaroya

Instrumentation and ControISystansBranch

Reviews and evaluates design, fabrication, and operation of reactor protection and safety instrumentation, and control instrumentation; participates in development of guides and regulations pertaining to instrumentation and control systems.

Chief — Rodney Satterfield

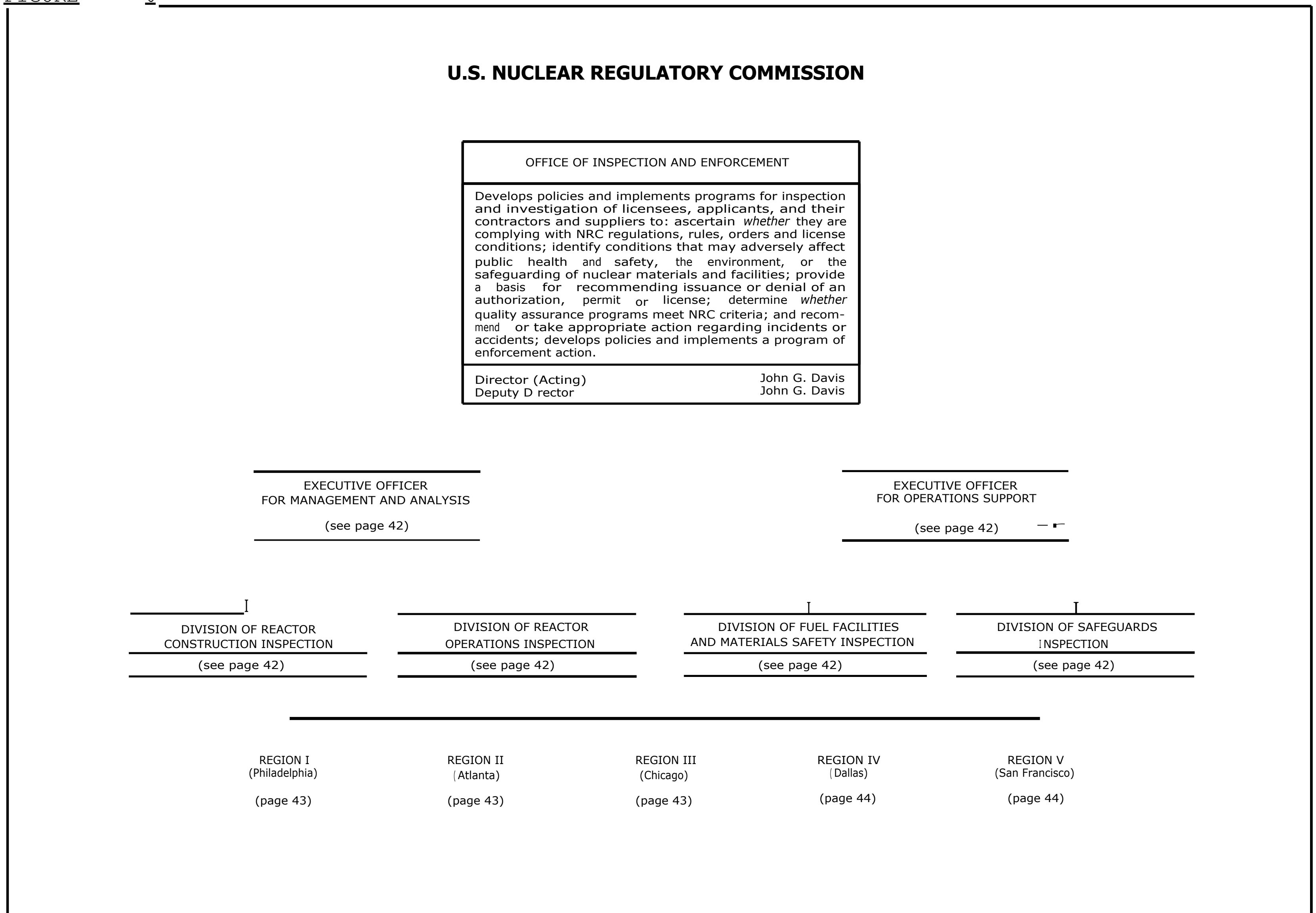
PowmrSVnanIBrarch

Reviews reactor license applications and related documents to evaluate the design, fabrication and operation of onsite and offsite electrical power systems and the steam and power conversion systems; participates in the development of guides and regulations pertaining to these systems.

Chief — Faust Rosa

Source: NUREG-0325, "U.S. Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.

FIGURE 6



Source: NUREG-0325, "U.S. Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.

FIGURE 7

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

Executive Officer For Management and Analysis

Administers functions of budgets, financial control, computer services, management information systems, planning, personnel management, contract administration, technology and inspection training, and management studies and analyses.

Leonard I Cobb

Executive Officer For Operations Support

Develops policy, criteria, and program requirements for enforcement and investigations; manages investigations assigned to Headquarters; assures that Headquarters enforcement decisions meet criteria; coordinates the enforcement program for consistency among the Divisions and Regional Offices; develops and administers the procedure and center for response to incidents; and provides centralized administrative support to the Headquarters staff.

Samuel E. Bryan

DIVISION OF REACTOR CONSTRUCTION INSPECTION

Conducts functions pertaining to operation of reactors. Develops the inspection program, assuring the technical adequacy of enforcement cases and investigations, preparing notifications to appropriate parties regarding incidents and generic issues, providing technical management and support to the NRC response to incidents, monitoring and appraising program performance by individual Regions and representing the Office to other NRC offices on matters of common interest.

Director Harold 0. Thornburg

DIVISION OF REACTOR OPERATIONS INSPECTION

Conducts functions pertaining to operation of reactors. Develops the inspection program, assuring the technical adequacy of enforcement cases and investigations, preparing notifications to appropriate parties regarding incidents and generic issues, providing technical management and support to the NRC response to incidents, monitoring and appraising program performance by individual Regions and representing the Office to other NRC offices on matters of common interest.

Director Norman C. Moseley

DIVISION OF FUEL FACILITIES AND MATERIALS SAFETY INSPECTION

Conducts functions pertaining to radiological and environmental protection at reactors, fuel facilities and in the handling of licensed materials, and for criticality control at fuel facilities. Develops the inspection program, assuring the technical adequacy of enforcement cases and investigations, preparing notifications to appropriate parties regarding incidents and generic issues, providing technical management and support to the NRC response to incidents, monitoring and appraising program performance by individual Regions and representing the Office to other NRC offices on matters of common interest.

Director James H Sniezek

DIVISION OF SAFEGUARDS INSPECTION

Conducts functions pertaining to protection of nuclear materials and reactors. Develops the inspection program, assuring the technical adequacy of enforcement cases and investigations, preparing notifications to appropriate parties regarding incidents and generic issues, providing technical management and support to the NRC response to incidents, monitoring and appraising program performance by individual Regions and representing the Office to other NRC offices on matters of common interest.

Director E. Morris Howard

Source: NUREG-0325, "U.S. Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.

FIGURE 8

U.S. NUCLEAR REGULATORY COMMISSION

DIVISION OF ENGINEERING STANDARDS

Plans and directs program for development of regulations, criteria, guides, standards and codes for nuclear safety in design, construction and operation of nuclear reactors, other production and utilization facilities, and facilities for the storage, processing and use of nuclear materials; and in materials safety activities, including production, use and transportation of radioactive products. Develops, reviews and monitors research and development programs to find solutions to safety problems related to nuclear reactors and fuel cycle facilities. Provides technical assistance on research and resolution of generic issues related to nuclear facilities, transportation of nuclear materials, or nuclear wastes. maintains liaison with other Federal *Agencies,* ANSI , international agencies, and other organi zations in assigned areas.

Director — Guy A. Arlotto

ASSISTANT DIRECTOR FOR GENERAL ENGINEERING STANDARDS

Plans and directs program for development of standards for safety in design, construction and operation of nuclear reactors, other production and utilization facilities, facilities for storage and processing of nuclear and radioactive materials, and other areas of nuclear safety. Prowiden advice and assistance on research and development programs and the resolution of generic issues involving safety problems of nuclear facilities.

Assistant Director — Wilbur M. Monsoon

svucnxu *and Components* Standards mends

Develops standards for design, construction and operalion of nuclear reactors and fuel cycle facilities with emphasis on mechanical engineering, structural engineering, and materials engineering aspects.

Chief — William F. Anderson

Reactor SritnnsStandmrdsBranch

Develops standards for export of nuclear reactors; design, construction and operation of nuclear reactors with emphasis on quality assurance, systems performance and design, instrumentation, personnel qualification and quality assurance for other fuel cycle facilities.

Chief (Acting) — Don Sullivan

Engineming Methodology Standards Brands

Develops standards for design, construction and operation of nuclear reactors and fuel cycle facilities with emphasis on methods of analysis and testing to assure specified performance characteristics of individual systems and the plant as a whole.

Chief — Awrma **A.** Norbrq

ASSISTANT DIRECTOR FOR MATERIALS SAFETY STANDARDS

Piano and directs program for development of standards for transportation of nuclear material, production or use of devices or products containing special nuclear material, Design and construction of fuel cycle facilities, storage and disposal of nuclear waste material, criticality safety, engineering aspects of occupational exposure, and chemical engineering aspects of nuclear reactors. Provides advice and assistance on research and development programs and the resolution of generic issues related to safety problems of nuclear facilities, transportation of nuclear materials, or nuclear wastes.

Assistant Director — Robert M. Bernero

TramportationandProduct StandrdiBrands

Develops standards applicable to the transportation of nuclear materials and to the production or use of devices and products containing source, byproduct, or special nuclear material.

Chief — Robert F Barker

Fuel Process Sysdams Standards Braids

Develops standards for design and construction of fuel cycle facilities; waste management, including processing, packaging, temporary storage and disposal of nuclear waste; decommissioning of reactors and fuel cycle facilities; chemical engineering aspects of nuclear reactors; criticality safety; and engineering aspects of occupational exposure.

Chief — Keith G. Steyer

Source: NUREG-0325, "U.S. **Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1,** Jan. 1, 1979.

FIGURE 9

U.S. NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS	
Reviews safety studies and applications for construction permits and facility operating licenses and makes reports thereon; advises the Commission with regard to hazards of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards; upon request of the Department of Energy (DOE) reviews and advises with regard to the hazards of DOE nuclear activities and facilities; reviews any generic issues or other matters referred to it by the Commission for advice. On its own initiative may conduct reviews of specific generic matters or nuclear facility safety-related items. Conducts studies of reactor safety research and prepares and submits annually to the U.S. Congress a report containing the results of such study.	
Chairman	Max W. Carbon
Vice-Chairman	Milton S. Plesset
Executive Director	Raymond F. Fraley
Asst. Executive Director	Marvin C. Gaske
Asst. Executive Director for Project Review	Morton W. Libarkin

Source: NUREG-0325, "U.S. Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.

FIGURE 10

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF MANAGEMENT AND PROGRAM ANALYSIS

Provides NRC staff with management information and program analyses; identifies and analyzes major NEC policy, program and management issues and conducts long- and short-range planning to assist NEC operating officials; develops and implements management Information and control systems and recommends policy on use of such systems for agency-wide applications; develops and implements application of sound statistical practices within NRC; and coordinates special information projects on overall NRC policies and programs.

Director — Norman H. Haller

DIVISION OFJNFORMATION. ANALYSIS AND PLANNING

Directs the establishment of design requirements for and the implementation of management information and control systems to support NRC needs; directs analyses of major program and management Issues; conducts short- and long-range planning as appropriate; provides advice and assistance to NRC components.

Director — Harold S. Bassett

Internal Information System Brach

Develops and implements management information and control systems pertinent to certain NRC activities; develops and issues Performance Appraisal Report for the EDO; issues Program Summary Report; administers and coordinates user requirements for NRC manpower system; develops, implements and refines decision unit tracking system.

Chief — Anthony F Abell

Special Pro*" Branch

Supervises preparation and publication of the NRC Annual Report to the President end Congress; coordinates the correction of transcripts of congressional testimony; coordinates and prepares in final form NRC response to congressional questions; prepares and edits special reports, speeches, briefings and other information issuances; administers management directives system and coordinates preparation of NRC organization charts.

C ief — William G. Dooly

Analysis and Planning Branch

Identifies and analyzes major regulatory issues and recommends agency options; assists EDO with policy studies in management and program areas; participates In formulating NRC planning assumptions for budget purposes and develops caseload-estimates; reviews value/impact analyses associated with regulatory actions; develops and maintains economic data bas* on nuclear industry; conducts long and short range planning activities.

Chief — Stephen Co river

DIVISION OF TECHNICAL SUPPORT

Directs collection, evaluation and publication of operating experience data from licensed nuclear facilities; design, development, and implementation of automated NRC management Information and control systems; and development and application of statistical methods and theory in support of NRC technical and administrative programs.

Director — Vacant

Automated Syaums Branch

Reviews and coordinates NRC requirements for auto.atic data processing equipment to support management information systems; administers interagency agreement for computer timesharing services; coordinates development and implementation of NRC management information and control systems and develops special studies in support of these systems.

Chief — Isaac A Kirk

Applied Statistics Branch

Develops and implements application of sound statistical practices in NRC technical and administrative programs, providing evaluation and technical assistance to NRC components in this area; conducts studies involving statistical techniques; reviws, prepares and presents testimony involving statistical technology; develops and maintains contacts with statistical community; provides educational activities in statistics for NRC staff.

Chief — Roger H. Noore

Licsnme Operations Eveluaion Branch

Reviews and transcribes reports for Licensee Event Report and Radiation Exposure Information and Reports systems; reviews and evaluates license* and NRC reports of unusual occurrences and incidents; prepares Quarterly Report to Congress on Abnormal Occurrences; prepares technical evaluations and special studies as required; prepares periodic status reports on licensed nuclear power plants; provides technical direction for the Nuclear Plant Reliability Data System.

Chief — Richard A. Hartfitld

Source: NUREG-0325, "U.S. Nuclear Regulatory Commission Functional Organization Charts," NRC, Revision 1, Jan. 1, 1979.

2. The responsibility to provide overall system engineering analysis requirements to ensure that a plant will operate as designed in all modes. No DSS organization is assigned system engineering responsibility according to Mattson (reference 12).

3. The responsibility to review the role of the operator or consider human factors in the design or design review process is defined in neither the DSS chart, Figure 5, nor do the DPM or DOR charts shown have this responsibility. This was also confirmed by Mattson in his deposition (reference 12).

4. The assignment to be a focal point to identify and to receive safety concerns and cause them to be reveiwed, resolved, and resolution applied.

b. Organizational Responsibilities That Appear To Leave "Holes"

1. The Quality Assurance Branch appears isolated. It should routinely provide requirements to the designers and reminders to NRC system reviews, to make sure the quality assurance requirements of 10 CFR 50, Appendix B, are properly imposed by designers and the utility. They should work closely with the NRC's I&E to accomplish QAB's stated responsibility of assuring continued compliance during plant operation.

In practice, the strength of the intended role of quality assurance in the NRC activity and in what NRC expects of the utility quality assurance function is questionable. QAB and QAB supervisors to the director level were not represented on the NRC Lessons Learned Task Force (reference 25), and I&E did not interview any Met Ed quality assurance people in the 200-plus interviews held during their investigation of TMI-2 accident (reference 52). Separating quality standards, quality engineering, and inspection appears to weaken the contribution of the discipline.

2. The functional charts depict I&E as being relatively autonomous from the project and design review groups. This is confirmed by the I&E manual and various interviews and depositions. This split keeps designers and project managers away from experience and I&E out of policy development.

3. Division of Operating Reactors appears insulated from most of the functions of licensing conducted by Division of Project Management and by DSS. There is a need to provide operating experience to the licensing effort and a need to provide the licensing experience to those who will monitor the performance of the operating plants.

4. There appears to be no internal technical audit function in NRC. The I&E in Washington, D.C. does review the activities of its inspectors, but there does not appear to be any organization responsible for reviewing and auditing the overall utility overview process. The Office of Inspector and Audit appears to be a legal and administrative audit only, not involved in technical reviews.

c. Organizational Responsibilities That Appear To Be Redundant

From general information obtained during the review it appears that there are redundant engineering capabilities in I&E, DOR, and DSS. I&E has engineering groups to review plant problems they feel then can handle. DOR has engineering groups to look at the operating reactors, and DSS does the engineering review of new plants. Such redundancy can be wasteful of talent and can result in "the ball being dropped in the crack." The I&E, DSS review of the Davis-Besse incident on Sept. 24, 1977, is a good example of this and is discussed elsewhere in this report.

Findings

- There is no assignment within the NRC organization for overview of critical functions such as problem reporting, failure analysis, and corrective action; systems engineering; and the role of the operator and human factors in plant safety.

- The fragmenting of quality assurance responsibilities among the various NRC organizations weakens the ability of this discipline to ensure an adequate quality program.

- The NRR Division of Operating Reactors responsible for overviewing the operating reactor is not part of the licensing design review, construction, or startup monitoring process.

- No NRC organization is identified as being responsible for auditing the project management, engineering, and inspection functions of the NRC.

2. <u>Evaluation of DPM and DOR in the Overview Process</u>

NRC responsibility for TMI-2 at the time of the accident was assigned to Light Water Reactor Branch No. 4, in the Division of Project Management. Harley Silver was the cognizant project manager. In the Division of Operating Reactors, Operating Reactors Branch No. 4, Jerry Zwetzry was the project manager for TMI-1. At the time of the accident, transfer of responsibility for TMI-2 from DPM to DOR was in progress. Also, Zwetzry was being transferred to Region V and it was expected that Domonic Dilanni of DOR would pick up both TMI-1 and TMI-2. The transfer process from DPM to DOR normally takes place 6 months to one year after licensing, but was being delayed somewhat, apparently due to the priority of other work.

To review and assess the activities of DPM and DOR, interviews were held with Robert Reid, chief, DOR Operating Reactors Branch No. 4, Harley Silver, Les Rubenstein, currently acting branch chief of DPM Light Water Reactor Branch No. 4, Donald Skovholt, DPM assistant director for quality assurance and operations, and various members of the DPM and DOR staffs. (These discussions are summarized in references 27-30.)

In addition, depositions given by Silver, Skovholt, Eisenhut, Rogers Boyd, and Bryan Grimes were reviewed. (These are listed under references 8, 9, 31, 32, 33.) These discussions, plus a review of the depositions, all provide the same picture of the role of the project manager, both in DPM and DOR, as being one associated primarily with the administrative control of the various licensing and review processes. The project manager does not appear to take a significant role in overviewing the overall management of the utility, the engineering aspects of the plant, or the operation of the plant.

Silver indicated he does not do in-depth technical reviews himself, but that all that activity was handled by technical specialists, primarily in DSS. Ross, in his deposition, also indicated the project manager is not expected to have a deep technical understanding of the plant. Boyd in his deposition confirmed the project manager's primary responsibility during design, construction, and startup was to assure that the review was conducted in an organized fashion. He conducted the necessary liaison between parties, provided meeting coordination and assured all questions asked were responded to. He apparently did not take significant part in assuring the quality of the question or quality of the answer although the NRC manager's handbook (reference 34) describes the project manager as needing to have both technical and managerial skills. It also states he must be capable of reviewing and understanding the efforts of others in highly specialized areas, developing comments and questions in regard to design criteria and design features, leading technical discussions, formulating overall technical discussions, formulating overall technical judgments, and writing engineering reports. In fact, the discussions with the project managers and their supervisors indicated that very little technical decision-making was vested in the project manager (references 28, 29).

The Project Managers Handbook primarily addresses itself to the administrative procedures associated with conducting construction permit and operating license activities. It gives the general indication also that the activity is conducted within the framework of the FSAR and Standard Review Plan (SRP) and appears to give little guidance as to whether the project manager should have significant knowledge of the operating plant. Confirming this observation, discussions with Silver and Reid indicate the project managers do spend most of their time in the handling of the various paper that constitutes the licensing and licensing amendment review process. Reid estimated that approximately half of a project manager's time was involved in handling license amendments in DOR.

No attempt was made to evaluate the project managers activities during the construction and initial licensing phase. The review primarily concerned his actions since licensing, during startup, and during the turnover and operation of the Three Mile Island plant.

Although DPM handles the facility through initial startup, and DOR follows the operating reactor, discussions with the two organizations indicate that the role of the project manager and the depth to which he reviews the overall management, engineering, and operations is very similar, and that the weaknesses of their review are applicable to both activities. The lack of in-depth knowledge of the facility is further degraded by the events of the turnover activities itself, particularly for the TMI-2 facility where a new project manager was to be assigned in DOR to both TMI-1 and TMI-2. The personnel interview indicated that normally DOR does not participate in the design review or licensing process of a reactor that they will be responsible for, so any general knowledge of a particluar facility maintained by the project managers is lost when the facility is transferred. The Project Managers Handbook states that a project manager must be the focus of information for the project assigned to him and that he should be more knowledgeable about the total aspects of the individual project than any other person (reference 34). A review of what the project manager does and how he does it paints an entirely different picture, and shows that the key person in NRC who is supposed to understand the particular site and how it is operated, in fact has little specific knowledge, and what knowledge there is in DPM is lost at the time of transfer to DOR.

A prime method for understanding the management organization of an operating facility is to conduct periodic visits to that facility to become well acquainted with the people, to understand and be familiar with the physical plant, and to observe and be knowledgeable of how the plant is being operated. Silver indicated he had visited TMI approximately twice a year from May 1975 through February 1978, and in our interview indicated he spent possibly 12 days on-site during the previous 2 years. Informal discussions with the project manager for Davis-Besse and the Arkansas nuclear plant indicated that although he had taken over responsibility for Davis-Besse in November 1978 and responsiblitity for the Arkansas plant in April 1978, he had yet to visit these two facilities.

Discussions with the aforementioned personnel, and reviews of the depositions provide a number of examples that indicate the lack of depth of knowledge that the project manager has, or is expected to have, relative to the engineering aspects and operation of the utility. In review of LERs from the utility it was generally stated that the project manager does receive all LERs concerning his facility and does a general review and scanning of those documents. However, it was stated that the project manager relies on I&E to be primarily responsible for the review and close out of LERs. In fact, the Project Manager's Handbook on pages 273 indicates that the project mnager is responsible for LERs "to extent referred to NRR by I&E, review and evaluation (of LERs) is the responsibility of the assigned project manager" (reference 34). As an example, Silver indicated that he, in fact, was not part of the discussion on close out of the incident at TMI on March 29, 1978, which involved a fail to open of the PORV due to loss of electrical power. Silver also indicated that he generally did not review the facilities' other nonconformance reporting systems used to report and evaluate problems that were not considred reportable to the NRC.

Silver also indicated that he was not involved in the review or approval of operating, maintenance, and other procedures at TMI. He indicated that this was included in the I&E inspection effort. Apparently, the prmary responsibility of NRR during the licensing and startup phase is to assure that the utility has the proper list of procedures and the organization to prepare, review, and approve them in accordance with the applicable NRC regulations.

All groups contacted generally indicated their overview activities did not include looking at the non-safety-related items that are not considered or covered by the FSAR or other NRC requirements. Apparently, the project managers do not use knowledge of these systems to give them some general feel for the overall "health" of the plant and the overall management capability of the utility's management. The personnel contacted indicated they relied primarily on feedback from the I&E inspection process and primarily on things that I&E brings to their attention. However, since I&E primarily reviews and considers safety-related and licensee technical specification items, activities outside of the scope of these activities are not available to the project manager for use in overall assessment.

With regard to system engineering and system interaction aspects of the utility plant, both DPM and DOR personnel agreed that the project managers do Not generally perform the role of system engineer to review system interaction problems. This confirms further that the project manager does not serve a significant engineering review role in the licensing effort.

Although the project manager in DOR is not a significant part of the licensing design review process, plant startup, or closeout of LERs, the project manager is expected to make significant decisions regarding changes to that facility that require changes to the license or technical specifications and therefore affect the safety of the facility.

Proposed changes are submitted initially to the project manager. He has the responsibility to review these changes and to decide what additional technical review is required. His supervision includes up to, the division director, who signs the approval of the change, but there is no independent review of this action, outside of line supervision, to assure proper review has taken place and that all necessary safety questions -- questions related to operator training, procedural activities, etc. -- have been taken into account. Discussion with project managers indicate that the changes are handled primarily within the scope of the FSAR and Standard Review Plan and they normally do not evaluate whether procedures or operator training should be changed.

The Quality Assurance Branch within DPM is primarily responsible for reviewing submitted quality assurance plans and Section 17 of the FSAR. They also participate in some of the startup tests to assure the tests are conducted in accordance with requirements. QAB personnel also indicated they rely on IE for feedback on any inadequacies in the quality assurance program. Once the facility is operating, the quality personnel indicated feedback data from IE was primarily used to see whether requirements should be changed or should the problem be considered in light of forthcomming license review activities. Little review of how well the contractor was fulfilling his job was evident. The QAB personnel also indicated one weakness in the system was that they approved the original quality assurance plan, but are not required to approve changes thereto. It is noted that changes of organizational structures which might affect quality assurance are part of the FSAR submittal and if the FSAR is changed these are reviewed and concurred in by the quality assurance group.

In summary it appears that the program management, quality, and supporting organizations within DPM and DOR conduct their review and overview activities within the scope of the FSAR and the SRP. However, as shown, this overview does not provide the project manager, or his supervisors up through the division director, a thorough understanding of the utility, its management, or the ability of that management to conduct its operation.

Findings

- NRC project managers and quality assurance personnel in the NRC Division of Project Management and Operating Reactors are primarily concerned with initial licensing and changes thereto within the scope of the FSAR and SRP. Little overall a assessment of utility management, engineering, or operations is evident.

- The NRC project manager does little engineering analysis and is not a significant factor in the review of nonconformances, procedures, or system engineering aspects of the plant.

- Project management experience gained during design construction and startup of the plant is lost upon transfer of responsibility for the plant to DOR. There appears to be little effort by the project manager in DPM to transfer licensing and startup experience to other NRC groups.

- There is no NRR review of proposed operating procedures as part of operating license approval.

3. Evaluation of Division of Systems Safety in the Overview Process

Although this paper primarily concerns the NRC and utility overview processes as they relate to the operating nuclear power plant, the role of DSS, which is primarily in the licensing process, had to be considered. As shown in this and subsequent sections of this report, the decisions DSS made during the FSAR review and the extent to which they utilized or evaluated experience from operating reactors had a significant part in the events or lack of events that contributed to the accident at TMI. The staff review included discussions with personnel from Division of System Safety and other NRC organizations interfacing with the division; reviews of depositions by personnel in the division or supervising activities of the division; and reviews of documents associated with the activities of the division. (The various meetings, depositions, and documents are delineated in references 2, 11, 12, 16, 18, 27, 35-43, 115, 116.)

As stated previously, the Standard Review Plan (reference 2) is currently the primary guide for DSS to conduct their review of the FSAR. The SRP was reviewed to provide insight into the depth of the review by DSS and what specific elements are investigated. As defined in its introduction, the principal purpose of the SRP is to assure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope of and requirements for the review process. Although the Standard Review Plan was initiated in late 1975 at about the time of the review of the operating license for TMI-2, and was not directly utilized in that review, Mattson in his deposition (reference 12) did indicate that the basic guidelines of the SRP were used in the TMI-2 review, but that TMI-2 was "grandfathered for a number of requirements in the SRP." A detail review of the plan would probably indicate there were a number of differences, but for the staff study the SPR is more rigorous than the review required for TMI-2. Any use of it in our investigation that shows potential weaknesses is applicable.

The SRP primarily addresses the design review of individual components and subsystems. No specifc section covers systems integration or systems interaction. On page 19 of his deposition (reference 35), Roger Boyd indicated "system interaction is not an integral part of overall review" and also "system (lack of system engineering) is one of the ongoing, so-called unresolved safety questions."

Mattson in his deposition (reference 12) indicated that in 1975 when the Standard Review Plan was assembled, NRC felt that if all the subsystems were reviewed in accordance with this plan, the systems would essentially play together. The individual sections that discuss the requirement for design review appear to generally ignore the requirements of the operator or the effect the operator can have on the system. For example, Section 7.3 and 7.4 of the SRP for engineered safety feature systems required for shutdown neither speaks to how these systems are to be operated, nor addresses requirements for reviewing interfaces with the operator or human engineering aspects. There are no requirements for control room design in the SRP, and DSS personnel confirmed that NRC does not look at control room design, control room layouts, or the general human engineering aspects of the overall control system design.

Section 7.5 of the Standard Review Plan does speak to the safety-related display instrumentation, and shows for example, that in paragraph III, item 4, under review procedures, "the control panel drawings are examined to determine the maximum separation distance between redundant equipment and circuits internal to the control boards in accordance with . . ." No discussion of operability human factor type information is required.

Review of the Safety Evaluation Report for Section 7.5 (reference 36) indicates that this instrumentation had been reviewed and that the proper instrumentation was displayed. Primary concerns were whether the instrumentation was qualified, whether there was physical separation of cable trays, etc. The DSS review evaluated design, not how it was to be used.

With respect to NRC overview of non-safety-related systems, the review of the SRP and the TMI-2 Safety Evaluation Report indicates that DSS reviews the non-safety-related systems only to the extent necessary to assure these systems are not required for plant safety and to assure that failures of these systems would not impair the capability of protection systems or safety-related systems to prevent or mitigate the consequences of the accident, or cause plant conditions more severe than those for which plants safety systems are designed.

In Section 7.7 of the TMI-2 SER, which applies to control systems not required for safety, NRC simply states that it agrees that certain control systems are not required for safety, that their design is similar to those of previously licensed plants, and that the applicant has stated he has not taken credit for their functions which might be available to help prevent an accident. The DSS concludes that differences in the detailed design (compared to previous plants) are minor and that the designs of these systems are acceptable for TMI-2. It is interesting to note that this type of review philosophy is applied to such relatively critical systems as the integrated control system and control rod drive control system.

With respect to mechanical systems that are non-safety-related, the DSS does do some review. However, the review is quite limited and generally associated with safety-related systems or structures in one sense or the other. For example, Section 10.4.6 of the SRP and the TMI-2 SER which applies to the condensate cleanup system (otherwise known as the condensate polisher at TMI-2) primarily address the fact that the structure meets certain structural criteria related to breaks in the high and moderate energy piping system outside the containment, provides assurance that there are no-safety-related components or systems located adjacent to its pipes, and meets certain other cleaning capability requirements related to conductivity. No overall design review is required nor is a failure mode and effects analysis (FMEA) conducted to ascertain what failure modes might cause plant to shut down or exercise the plant emergency feedwater systems. The TMI-2 SER appears to indicate that the DSS review of these particular systems and the overall condensate circulating water system was primarily concerned with failures in the circulating water system which could result in flooding the turbine building basement where there was a common wall between the turbine and the control building area which was of seismic category 1 design. There is no evidence that DSS did any significant review of the TMI-2 polisher from a performance capability point of view. Discussions with Novak (reference 27) and the deposition of Mattson (reference 12) also confirm that DSS review was primarily associated with the safety-related systems and relatively little attention was given the non-safety-related hardware in the facility.

In the area of operating procedures of all types, the SRP, the SER, and the various interviews and depositions all confirm that the review process is primarily concerned with assuring the utility has the proper management structure and systems in place to prepare and review procedures. In addition, the SRP requires that the proper list of procedures and types of procedures will be in place and that there is an independent review process planned by the utility. A specific list of procedure categories is included in the SRP. These documents also confirm that DSS does not review any actual operating procedures but relies on I&E review to assure the procedures meet basic requirements. They agree that there is no integrating of the operating procedure review with the design review. Mattson in his deposition (reference 12) indicated this was probably a mistake which should be corrected in the future.

The SRP and the DSS organization structure appear to be significant factors in the narrow focusing of DSS personnel attention in their review of reported incidents that kept them from being able to recognize the implications of various signals and clues that the overall system and operator might be in trouble. However, their response to a particular event at Davis-Besse on Sept. 24, 1977, is illustrative of the extent to which the division, and NRC in general, focus their efforts and thinking on components and subsystems and apparently did not think in terms of systems interactions, operator interactions, operator problems, and operating procedures (reference 12). This incident is a prime example of the thinking and review process that was occurring at NRC prior to the Three Mile Island accident.

A transient occurred at Davis-Besse-1 on Sept. 24, 1977, that had significant characteristics in common with the TMI-2 accident. The transient included a stuck open PORV, rising pressurizer level, early operator throttling of high pressure injection (HPI) during the transient, etc. The event was reported to the NRC Region III Office in LER NP-32-77-16 (reference 117). The details of the event are discussed in other staff reports. The incident was of enough significance to initiate action by both the Office of Inspection and Enforcement and the Office of Nuclear Reactor Regulation. Gerald Mazetis, a reactor systems engineer from DSS, headed a small team of NRC personnel who met with representatives of Toledo Edison, Babcock & Wilcox, Region III office, and the Bechtel Corporation at Toledo Edison on Sept. 30, 1977, to discuss and review the overall incident. A group of approximately 32 people, representing a broad spectrum of knowledge relative to the facility operation, attended this meeting. The results of the meeting are documented in a trip report from Mazetis which he prepared upon his return to Bethesda, Md. (reference 37). In this report Mazetis does state that the operator secured the HPI pumps because he observed a restoration of increasing pressurizer level. But Mazetis does not relate that particular activity to any abnormality of operation nor does he consider it a particular problem. His primary concerns, which show the general emphasis by DSS on hardware, were associated with items such as the affect of excessive cooldown rate on the primary side, and stresses in steam generator-2 which apparently went dry. He did note the vapor formation in the reactor coolant system, but his concern was related to the effect of this vapor on reactor coolant pump seals; again worrying about the mechanics of the reactor cooldown pumps rather than on the overall systems operation.

According to the depositions of Mattson, Ross, and Mazetis (references 12, 16,38), Mattson and Ross received copies of this trip report, and a meeting was held in Mattson's office on Monday, Oct. 3, 1977, to review the transient. According to Mattson in his deposition, the meeting was attended by a number of key DSS and I&E personnel including Karl Seyfrit, who was at the time a senior manager in the Office of Inspection and Enforcement, Tom Novak, a branch chief in DSS, Mazetis, Mattson, and probably Ross. Mattson indicated that this meeting lasted several hours and reviewed Mazetis' trip report, but also had a primary objective of determining whether the Office of Inspection and Enforcement or the Division of System Safety was going to continue the investigation.

The DSS' primary concern was that I&E consider the specific DSS concerns and assure an adequate closeout and corrective action.

In spite of the fact that there was a meeting of 32 experts in Toledo and a meeting of the available expertise at NRC on this particular incident, the whole attitude of all concerned was apparently so ingrained in thinking in terms of component and subsystem design and performance aspects that no overall realization of the system interaction/operator interaction problems of this incident was recognized (reference 12). Ross, assistant director for reactor safety in DSS, did send a note to Karl Seyfrit on Oct. 20, 1977, (reference 39) regarding the Davis-Besse occurrence and stated four particular areas of interest that DSS felt were appropriate for the final Toledo Edison report on this incident. These included the areas previously noted by Mazetis in his trip report, but also included an item regarding the operator's role in the event and his decision to secure HPI flow based on pressurizer level indication.

According to Mattson, there was no further followup between I&E and DSS on this particular incident. It is interesting to note that in the October 7 letter to Region III from Toledo Edison, providing additional supplemental information on the LER (reference 40), that no mention of operator action or pressurizer level going high is discussed. In reference 41, Toledo Edison's final 90-day report for this event, dated Nov. 14, 1977, had no discussion of possible early operator action in termination or throttling HPI during the event; no information is provided relative to corrective actions associated with procedures, and there is no recognition or discussion of the fact that a system condition existed which had not been previously analyzed or for which the operators had not been given training to recognize and react to. The report primarily discusses the various equipment failures and the actions taken by Toledo Edison in response to these failures. It is interesting to note that the analysis of the reactor coolant system section of this report primarily concerns items such as stresses in the pressure boundary, fatigue life of the reactor coolant components, the affects of the changes in water level on pressurizer shell stresses, thermal shock to the heaters, etc. This is quite indicative of the effect that NRC regulations have on the utility in that the utility tends to discuss those things they feel the NRC is primarily interested in. With respect to operator actions and training, the report simply states that the operator action was timely and proper throughout the sequence of events, and that a training program was developed and presented.

NRC I&E inspection report number 50-346/77-32 was transmitted to Toledo Edison on Nov. 22, 1977, from Region III (reference 42) and is apparently the closeout of I&E review of this incident. A review of that document also indicates that the inspector was primarily concerned with the hardware failures and corrective actions associated with this hardware. There is no evidence of any discussion relative to operator actions or a recognition that operator/systems interaction had been considered or were being considered in corrective action. Again there was a brief reference to a training program conducted by Toledo Edison.

It appears from the above and confirmed by the depositions of Mattson, Novak, Boyd, and others that the Division of System Safety would not have recognized or reacted strongly to indications from I&E regarding operator and system level problems because of their primary interest in the design of components and subsystems, particularly as this data related to their next licensing action. Even Novak, who by Jan. 10, 1978, in the famous Novak memorandum to Reactor Systems Branch members (reference 43) was beginning to recognize the potential for erroneous pressure level readings and the effect that these might have on operator procedures, apparently was not concerned enough to send such concerns to personnel directly involved in the overview of plant operation, but primarily was interested in alerting his branch members to this potential problem in any forthcoming construction permit or utility operating license reviews.

Further review of the various reference documents indicates that the Division of System Safety receives relatively little input or uses relatively little input regarding operating experience in order to determine changes that should be made to the various design criteria or to the review processes. They basically rely on I&E coming to them when there are problems or information that I&E feels that might be pertinent to DSS. As discussed in Section III-D, there is no systematic failure closeout or overall operating review activity that involves the project management, engineering, and inspection sides of NRC. Both Boyd and Mattson in their depositions commented that NRC was not tracking equipment failure history or operating experience in order to modify or change NRC requirements, or how NRC was doing business. Boyd in his deposition (reference 32) indicates that he knew of no section of NRC responsible for relating operating experience gained from the various reactors to particular requirements or regulations. One reason for this apparently was due to the fact that in all safety analyses NRC had essentially ignored the operator. They knew the operator could contribute to preventing the accident, but did not assume any help from the operator in their failure analysis. Conversely, they did not consider him to be part of the failure sequence either.

In summary, the staff review has indicated that the Division of System Safety is the engineering and technical arm of NRC. They are primarily concerned with the review of applications for construction and operation of nuclear power plants. Their efforts are applied primarily to the design and performance of components and subsystems that are considered safety-related, and little attention is given to other plant systems, the role of the operator, or the experience the operating plants are providing.

Findings

- The Division of System Safety overview of the nuclear power plant is primarily concerned with the design of safety-related components and subsystems within the framework of the Standard Review Plan.

- The DSS does not include nor does the Standard Review Plan require significant consideration of non-safety related systems, total systems interactions, operating procedures, or human factors in the evaluation of the nuclear power plant.

- The DSS has not adequately recognized potential system and system-operator problems even when these problems were brought to their attention; -- possibly because of the emphasis applied to component and subsystem design aspects and to the design-basis accidents by the NRC.

- The DSS makes little use of plant experience data in developing requirements for and in the conduct of their overview process.

4. Evaluation of the Office of Inspection and Enforcement in the Overview Process

a. Background

The NRC Office of Inspection and Enforcement has the primary role in the NRC overview of the operating utility. Its regional offices are the primary interface between the utility and NRC and are the "eyes and ears" of NRC in the determination as to whether the utility is meeting its commitments as defined in the FSAR, operating license and technical specifications. All knowledge that defines how well the utility is doing its job, except for utility proposed license changes, flows through the regional offices to I&E in Washington, D.C., and hence to the rest of NRC. In many critical areas I&E decides if the engineering and project management parts of NRC -- DSS, DPM, and DOR should be involved in a particular action. For the majority of the day-to-day decisions the NRC makes regarding the acceptability of utility actions, the I&E regional office is sole adjudicator.

Scope. To evaluate the adequacy of this overview process, meetings were held with representatives of the Region I Office and personnel from the Division of Reactor Operations Inspection, in Washington, D.C. Depositions of key personnel in the I&E organization and from other NRC groups that interface with I&E were reviewed. I&E Manual chapters relating to the I&E operating reactor inspection process were evaluated and various I&E inspection reports and other documents and correspondence were reviewed. (These visits, depositions, and documents are summarized in references 17, 18, 44, 45, 46, 49, 52-55, 62, 63, 80.)

The role of I&E in overview of utility procedures, configuration control, and nonconformance reporting is discussed in detail in other sections of this paper as is the basic organizational structure NRC has assembled to conduct this overview. This section primarily addresses I&E's overall performance of what they do and how they do it as they overview the operating reactor.

b. The I&E Manual

The NRC Inspection and Enforcement Manual (reference 44) is the primary controlling document for I&E's overview process. It spells out in detail what is done by I&E, what is done by the inspector, what meetings are held, how reviews are to be conducted, how often they are conducted, and provides specific check sheet procedures to be followed when the inspector conducts a particular audit. Chapter 2510 defines the basic overview process for power reactors and chapters 2514 and 2515 apply to light water reactor startup and operation phases. For those areas covered by the I&E program, the manual appears to document a complete, detailed, and well-written audit program to a degree and volume not observed even on other programs (DOD and NASA). It is noted, however, that a recent study indicated that I&E inspectors found the manual difficult to follow.

In spite of a well planned and documented program, the I&E overview process has serious deficiencies that deny the NRC a true understanding of the "health" of the operating facility and the ability of the utility management to direct the operation of the plant. These deficiencies involve a number of factors including:

- the failure of the inspector to become involved in detecting and evaluating signals and indications of potential problems that are not strictly covered by the FSAR, technical specifications, license, and detailed requirements of the I&E Manual;

- lack of physical inspection activity on the part of the NRC inspector as opposed to the review of paper, reports of inspections, review meetings, or operations, etc.;

- lack of sufficient personnel to assure in-depth, more frequent overview; and

- no significant NRC I&E management to utility management link which would provide I&E with both an assessment of utility management capabilities and concerns, and provide both the I&E and utility management with an independent look at what the inspector is saying about the plant and its operation.

These deficiencies, coupled with I&E deficiencies in procedure review, nonconformance reporting and evaluation, and change control (discussed in sections III-C, D, and E of this report) are believed to seriously hamper I&E's ability to assure a safe plant.

Reviews of the Inspection and Enforcement Manual, various interviews, depositions, and inspection reports provide a good look at what the inspector is supposed to do, what he actually does, and how he does it.

Inspection and Enforcement Manual chapter (MC) 2500 describes the elements that went into developing detailed inspection requirements, from regulatory requirements, regulatory guides, industry standards, etc. It indicates that the inspection requirements contained in the manual may be in conflict with NRC requirements placed on a specific licensee and in that case license requirements take precedence. In this light, the inspectors agree that the primary guides are the FSAR, technical specifications, and operating license. Chapter 2500, however, does say "The I&E inspector is not limited to inspecting only those activities related to current regulatory requirements or specific license commitments, but in addition, he shoulders responsibility for identifying areas wherein the licensee is not meeting its responsibility to construct and operate the facility safely" (reference 44). The latter responsibility is reflected in numerous inspection procedures.

c. Inspector Action

Out review indicates, however, that at least for TMI-2 the inspectors of Region I primarily addressed themselves to the FSAR and technical specification items -- i.e., concentrated on safety-related items -- and did not accumulate or communicate a set of knowledge that would have permitted them to identify "areas wherein the licenses is not meeting its responsibility to construct and operate the facility safely."

Don Haverkamp is the project or principal NRC inspector for TMI and, as defined by Moseley in his deposition (reference 45), has responsibility to both conduct and coordinate the inspections of TMI-1 and TMI-2. Thus he conducts a number of the inspections himself, reviews the inspection reports on TMI done by others, and coordinates and schedules the overall inspection program at TMI. As Moseley indicated, and confirmed by Haverkamp, the project inspector is not a supervisor of the other inspectors and apparently does not approve their reports, but is aware of the data contained therein and must followup on any open items from the particular inspection. The region keeps a computerized outstanding items report against each facility by inspection report number and item number. This logs all deficiencies, infractions, deviations, etc. noted by the inspectors and shows on which inspection report the item was closed out. A number of inspection reports for the period from March 1978 to April 1979 were reviewed. The reports are well-written and apparently document all the inspector's activities conducted in support of TMI-1 and TMI-2 and the current status of open utility actions on previous inspection findings.

The report documents what the inspector does at the home office in addition to what he does at the utility. Unfortunately, the rigorous planning, reporting, and computer control of all activities documents a number of weaknesses in the system. Some examples -- "Inspection Report 50-289/79-01; 50-320/79-01 Transmitted to Met Ed on January 29, 1979," (reference 46). The inspection was conducted by Haverkamp, approved by D. Johnson for R. Koinig, chief reactor projects section no. 1, and the letter of transmittal was signed by Eldon Brunner, chief, Reactor Operations and Nuclear Support Branch. Haverkamp has conducted an unannounced plant operations inspection, an inspection which is required quarterly by I&E Manual procedure 7170B. The inspector stayed 4 hours at TMI-1, 16 hours at TMI-2, and 8 hours at Reading, Pa. Seven previously unresolved items were closed out. A number of them were closed without the inspector seeing the final licensee action; licensee still proposing options to NRR to correct a problem with low water in the steam generators at low power level; permanent revision of SAP1002 (requiring selection and testing of emergency shutdown equipment) still in progress; technical specification change request submitted to NRR, but not yet approved.

A relatively thorough review of the paper associated with licensee reponse to an I&E circular was done at Reading, Pa., though no actual inspection of hardware changes took place although the circular appeared to require a number of plant modifications. The primary purpose of the inspection, an audit of plant operations, was conducted and included tours of the plant and review of various logs. The report indicates "control room staffing and control board monitoring instrumentation and equipment were observed for conformance with applicable technical specification requirements." The report describes the tours and appears to say the inspector observed housekeeping, fire prevention, proper posting of lockout tags, reasons for annunciators being on, etc. No noncompliance was noted.

The report shows, and Haverkamp confirmed, that no procedures or procedural changes were reviewed, nor had surveillance tests or other specific operations been observed other than normal control room operations. In his deposition (reference 17), Haverkamp confirmed that he had never been in the control room during a startup or shutdown nor had he attended a Plant Operating Review Committee (PORC), Generation Review Committee (GRC), or General Office Review Board (GORB) meeting. He indicated they (NRC) only looked at 5 to 10 operating procedures a year, less than 5 percent. It should be noted here that another reactor inspector had done a review of procedures as part of the Region I startup review on May 30 to June 2, 1978. (This inspection report, 50-289/78-12, 50-320/78-22, is discussed in section III-C). When asked about instruments being out of calibration in the control room, such as the radiation monitoring instruments noted by the technical staff during a tour in July 1979, Haverkamp indicated he noticed these, but since it was not a technical specification item, he normally would not comment on them.

He had not reviewed the EF-V-12A/B surveillance procedure which closed both valves after procedures were changed in August 1978, in violation of the technical specification, but was aware and essentially concurred in the destruction of the as-run surveillance procedures check lists after their completion. It is also noted that NRC did not detect the violation of the technical specification involving valve closure despite frequent inspection (I&E) visits to TMI-2, as evidenced by a summary of inspection reports (reference 80), which indicated 15 inspection periods with indication of few noncompliances found between August 1978 and March 1979. In addition, it is noted in this reference, that an I&E inspection was made at TMI-1 and TMI-2 on March 19-23, and March 26, 1979. Surveillance procedure 2303-M27A/B on the valves was last accomplished before the accident on March 26, 1979.

The technical staff in its tour of TMI-1 and TMI-2 had noted a number of poor industrial practices during their tours in May and July including poor housekeeping and poor marking of piping and valve systems. Haverkamp felt housekeeping was satisfactory, but "barely." He did not look at things like piping marking or the condensate polisher as these were not technical specification or safety-related items. He had not observed the PORV outlet temperature being high due to a poor history of high leakage, but felt the 130°F limit on this was a procedure limit, not in the technical specification so he probably would not have considered it. Apparently the inspector "observes" tags and alarms and assures they are properly noted and understood, but no assessment is made regarding whether the number of tags or alarms have some meaning or whether they obscured displays, indicators, or switches, as long as no technical specifications or requirements are violated.

Although certain inspections done by a specialist may be done well, such as I#50-289/78-20, which covers the calibration of safety-related components, one example, selected randomly from the computerized Region I Outstanding Items Report (reference 63) is very appropriate for further discussion. This report -- Inspection Report 50-289/78-16, 50-320/78-26, dated Aug. 14, 1978, (reference 50) -- was conducted by a radiation specialist. The report was approved by a Robert Bores for Stone, chief of the Environmental and Special Project Section. The transmittal letter was signed by Hilbert Crocker, acting chief of the Fuel Facility and Materials Safety Branch. Apparently, this lineup of supervisor review is standard practice for Region I. There is no evidence in the documentation that any one particular supervisor sees and approves all reports of a plant in order to get an overall picture of "how goes it" at the plant. The inspector conducted an emergency planning inspection, including looking at licensee off-site support coordination, emergency facilities, equipment, instrumentation and supplies, procedures, etc. The inspector appeared to conduct a thorough review citing the licensee for failure to have updated emergency plan implementing procedures in four emergency monitoring kits. He let pass, however, the fact that he had trouble "verifying that all persons assigned specific emergency duties and responsibilities had been trained as required by implementing procedure 1670.9, Emergency Training and Emergency Exercise, Revision 5 dated Feb. 15, 1978." He took the licensee's word that the first iteration of training under the new emergency procedures would start **in** August. He did talk to some employees and ascertained they had received training in August and September of 1977, prior to release of Revision 5, but he did not seem to worry about the apparent overall training deficiency.

The same inspector sampled calibration inventory records for equipment and took a standard mock iodine source to check four randomly selected emergency counting systems. The inspection at this point looked very thorough and complete. The results of this last step speaks for itself:

> Results obtained from using procedure 1670.6 and the selected field counting system were two orders of magnitude lower than the standard source activity. The check was repeated on each of the three remaining counting systems in order to determine if similar inconsistency would be evidenced in all of the system. Final results indicated that two of the four counting systems were within 12 percent of the standard source activities and two were in error by greater than 100 percent (one high, one low) of the standard source activities. Further review of the two systems which gave the inordinately high and low results revealed that one had faulty timing circuit and the other did not appear to be responding properly. Both faulty instuments were delivered to the instrument and control shop for repair.

The inspector noted that the two remaining systems were adequate in number to support an emergency response as described in the licensee's emergency procedures.

The inspector took no action to select other instruments to see how bad the problem was and no followup could be found in subsequent reports. It is noted that the I&E reports evaluating the TMI accident, found about half these instruments could not be used at the time of the accident (reference 52). In most inspection organizations, it would be normal for an inspector, after determing that half of the selected sample did not meet specification requirements, to have requested inspection of a much larger sample, or perhaps even the entire stock.

A review of a number of inspection reports for 1978 conducted during the startup phase of TMI-2 to confirm startup test results, indicates these reviews were well done. This activity is delineated in detail in the appropriate manual chapters associated with MC 2514. They appear rigorous and address I&E activities that ensure technical specification requirements and parameters are met. Although inspectors apparently do some independent review of test data during startup, and there are certain mandatory tests that NRC must observe, the general format for the inspection program is not a hands-on effort where the inspector participates in or repeats a physical inspection.

Although Region I management in our interview (reference 53), and Grier in his deposition (reference 54), implied a significant percentage of procedures are reviewed (25 percent) and the review includes step-by-step walk-through of procedures with drawings, Haverkamp's deposition and the inspection reports examined could not substantiate this. In addition, the IE inspection plan does not appear to include any provision for reinspection of hardware on an audit basis or similar techniques used by other government agencies such as Defense Contract Administrative Services (DCAS).

d. Quality Audit Activities

Review of the I&E inspection program and their actions until recently also show that I&E has minimal concern for quality assurance programatic aspects or the role of the quality assurance organization in independently assuring the safety and health of the plant. For example, MC 2515 indicates many of the quality assurance program audits are only required to be conducted every 3 years. Included are such critical areas as configuration control, procurement control, housekeeping/cleanliness, tests and measurement equipment, etc. Security activities, on the other hand, are generally on a semiannual review schedule. It is also interesting to note that I&E interviewed 203 people to investigate the TMI-2 accident as noted in the accident report, NUREG 0600. They did not appear to have interviewed one member of the quality assurance management or staff at TMI. Also, inspection reports are sent to a number of organizatonal segments at TMI, but not to the Quality Assurance Program.

Up to the time of our visit to the region in early July, Region I had not conducted a quality assurance audit at TMI. This apparently was not done as part of startup. A very complete quality assurance audit was run over a period of 3 weeks from July 17 to Aug. 2, 1979. Reference 55 is a draft report of this audit and clearly indicates that from the number of nonconformances found (eight) and from the number of inspector concerns that conditions existed that I&E should have reviewed and surfaced earlier. Specific details of the findings are discussed elsewhere in this paper.

e. Inspector Staffing

The staff review also observed some potential problems in the number of staff available to do an adequate review. Region I handles about 20 reactors plus all other general licensing in the northeast United States. There are about 145 people in the regional office of which 110 are technical. There has been some loss due to promotions and placing of personnel in the resident inspector program. Moseley indicated that a supplemental request is going in for additional unit inspectors for the sites, but a general theme of persons contacted was the lack of people or time to do the job required. Review of inspection report records indicates that the region does meet Grier's and Moseley's estimate that 20-30 visits are made each year to each site, with a total of about 1,000 staff hours spent at each site. Reviewing Haverkamp's deposition and the inspection reports confirms that although the inspectors meet the basic letter of the I&E program, there probably is little extra time to meet the general intent of the program or take on the necessary additional surveillance of the non-safety-related procedures and hardware.

f. Inspection Management

Our review above discusses the apparent lack of single manager in the region who has the responsibility for a general overview of the plant and the I&E inspection process to assure the plant is being operated safely. In a normal government overview of a contractor in a safety critical program, not only is there an overall government project manager to provide focus, but there also has developed a higher level government management-to-contractor management relationship that serves to overview both organizations ability to conduct the program and to provide independent assessment of the decisions being made at the working level. No such relationship appears in the NRC-utility overview process nor does I&E appear to evaluate the quality of the utility management particularly from a management-to-management viewpoint.

The I&E Manual does contain a number of procedures delineating requirements for periodic upper level region and corporate meetings during the construction, startup, and operating phases (Procedures 30001B, 30301B, 30700B, etc.). Procedure 30700B, issued July 11, 1977, defines the requirements for the third corporate management meeting to be held approximately 6 months after licensing and shortly after the startup testing program has been completed. This meeting is to involve senior corporate and region personnel and representatives of NRR. Apparently, this is to be a general meeting, but does discuss problems, general conduct of the licenses performance, and the transfer of the utility overview from DPM to DOR. Similar meetings are to be held every

3 years in accord with procedure 30702B (July 1, 1977) which includes I&E, but appears not to involve DOR.

According to Haverkamp's deposition (reference 17), a meeting was held in the Region I office on Feb. 9, 1979. The agenda and attendees at the meeting are contained in documents provided by Grier to the Commission (reference 57), as requested in Haverkamp's deposition. The agenda and list of attendees indicates neither DPM nor DOR were present or their items of discussion included. The meeting is quite general and does not appear to serve as a mechanism for management surveillance and review.

This staff review has concentrated primarily on NRC's I&E and its relationship to TMI. No detail assessment of I&E overall regional activity could be accomplished, but three inspection reports from Region III to Toledo Edison Company were examined as part of the review of the Davis-Besse-l Sept. 24, 1977 transient. These included IR50-346/77-31, 33, and 34 (reference 95). Review of these inspection reports tends to support some of the above findings.

Region III follows the same practice of having the inspector and his immediate line supervision approve the inspection report and sign the letter. No single overview activity appears in Region III either. Report 50-346/77-31 documents a Region III management meeting with Toledo Edison held at Davis-Besse on Oct. 27, 1977. The one-day meeting was attended by the region director and two representatives of DOR. An exchange of information took place regarding certain activities at I&E and the basic responsibilities of DOR and I&E and the turnover process from DPM to DOR. Although all necessary parties were involved in this meeting, it did not appear to be a strong vehicle for management assessment.

The other two Region III reports covered the normal inspection process. The reports appear to be more technically oriented than those written in Region I, generally containing some technical discussion of each item investigated. The reports do not, however, formally track each item of inspector effort, LER status, and unresolved items as do the Region I reports. This small sample indicates there may be differences in emphasis in the NRC inspection process among the NRC regions, but general weaknesses identified in Region I may also exist in other regions.

g. Region I Overview of Construction Phase

During the review of the preliminary draft of this document, "Quality Assurance and Reliability of Nuclear Power Plants," it was noted by one participant that the inspection reports initiated by the AEC during the construction phase of TMI-1 and TMI-2 contained a number of significant sounding deficiencies. It was decided that an assessment of the closeout of the deficiencies should be made by reviewing the closeout of the deficiencies noted in one particular inspection report which appeared to contain a majority of the deficiencies noted during the period 1970-74.

The deficiencies were those reported in an AEC inspection letter dated Oct. 6, 1972 (reference 127), which listed deficiencies found during inspections Aug. 14-17, 1972. There were 31 deficiencies cited in which some requirements were not satisfied in the "eyes" of the AEC inspectors. At the conclusion of the Aug. 14-17, 1972 inspection, these deficiencies were discussed by the inspectors with members of the Met Ed staff (reference 127) and were discussed in a telephone conversation between J. Miller of Met Ed and the AEC Region I director, as also noted in reference 127. In a Nov. 3, 1972 letter (reference 128), Met Ed responded to the deficiencies noted in the Aug. 14-17, 1972 inspection. On Nov. 9, 1972, the AEC held a management meeting review type of inspection with Met Ed management as noted in IR50-289/72-19 and 50-320/72-08, (reference 129), "to discuss with corporate management the Quality Assurance history of the Three Mile Island projects." Apparently, as a result of this inspection and other problems, a reorganization of quality assurance at the site did take place and additional support was provided by GPUSC (reference 132).

Later, the AEC, in IR50-289/73-02 and 50-320/73-02, (reference 130), after an inspection at Three Mile Island on March 26-28, 1973, provided in the "DETAIL" section of the inspection report, status information of the deficiencies noted during the Aug. 14-17, 1972, AEC inspection. Of the 31 deficiencies noted in August 1972, 12 were still classified as open, one had an unknown status, one was reopened by repeated infraction, and 17 were noted as closed. (This count may only be approximate because the status working, in some instances, failed to make it clear if the deficiencies were classed as "open" or "closed" and no attempt was made in this analysis to determine the supportability of the status information.)

Thus after about an 8-month period only slightly over half of the deficiencies were rated as closed by the AEC. Some of those remaining open at the time appear to have been significant enough to warrant emphasis for corrective action to bring about closure. Some examples of items open in March 1973 are as follows:

- assure controlled storage conditions are maintained on certain equipment;

- notification of quality control prior to installation of electrical engineered safeguard equipment; and

- lack of required separation distances and/or barriers between cable trays in safeguards actuating cabinets.

At the request of Commission staff, Region I searched available records and also found one inspector who had participated in this inspection and management meetings. Additional data provided by Region I (reference 131) indicated most and probably all items were closed out during management meetings on May 1 and 8, 1978, and by a Met Ed letter, May 15, 1973. Although the documents do not specifically define which unit the deficiences apply to, the NRC review indicates the audit was a midterm quality assurance audit of TMI-1 with the findings primarily applicable to that unit. It was noted that a few items applied to both units.

From this sample it appears that deficiencies found by AEC inspectors were not expeditiously pursued by the utility or by AEC to seek closure, but eventually were closed to the satisfaction of the region.

Findings

- The NRC Office of Inspection and Enforcement and its regional offices conduct a detailed, documented inspection program for those utility systems and activities covered by applicable regulations, regulatory guides, utility FSAR, operating license, and technical specifications.

- Region I on-site inspections appear to miss signals and symptoms that indicate potential plant operating problems and weak utility management.

- In Region I, there is little physical inspection or direct observations of operations such as surveillance testing of the operating reactors during NRC plant visits.

- Region I inspectors did not detect the emergency feedwater valve procedure change leading to technical specifications violation in about 15 visits to TMI-2 from August 1978 to March 1979.

- The role of quality assurance does not appear to be an important factor in the I&E plan. No I&E audit was made to TMI-2 quality assurance plan to see that plan was implemented to support the operating phase from the beginning. An I&E audit about 18 months after operating license issuance found many deficiencies in the implementation of the quality assurance plan. In their investigation of the TMI accident, I&E did not interview any Met Ed quality assurance personnel in the 200-plus interviews held.

- Sufficient I&E staff may not be available to conduct an adequate overall plant surveillance (inspection) activity.

- There is little I&E assessment of the utility's management capabilities.

- Although one inspector receives all reports concerning TMI-2, he has no responsibility for the execution or the quality of execution of all TMI-2 inspections.

5. <u>Evaluation of TMI Organization and Performance</u>

a. Scope-Background

The TMI organization, procedures, and practices were reviewed to determine whether they contributed to the accident at TMI-2 on March 28, 1979. This review primarily addressed the Met Ed management, engineering, operations, and assurance activities that were in place in response to NRC requirements for an operating reactor. The TMI facility had been licensed in February 1978 and had gone through its startup program through December 1978, at which time it was put into commercial operation.

During design, construction, and startup, the facility was under the direction of the General Public Utility Service Corporation. A separate and distinct quality organization existed during this period from that utilized when Met Ed took over responsibility for the facility. The Met Ed management team of operations, engineering, maintenance, and quality assurance, which was in place operating the TMI-1 facility, gradually assumed responsibility for the facility during 1977-78 as various portions of the facility were activated. Final acceptance of the overall facility was in the latter half of 1978. Since the design, construction, and startup organization no longer exists, the primary emphasis of the staff review was placed on the Met Ed organization as it existed at the time of the accident. However, a number of the procedures and practices which were in place during the startup phase were reviewed to determine if they were part of the events that led to the accident.

Time did not permit a detailed evaluation of all aspects of the TMI overview and quality assurance program. However, a general review of the organization, procedures, and practices was completed, and a detailed evaluation of the activities related to nonconformance reporting and corrective action, operating and maintenance procedures, and configuration control was conducted as these activities appeared to have had a direct bearing on the accident. It was felt that general quality assurance weaknesses found from such a review could probably be assumed to occur in other quality assurance functions such as procurement control, calibration control, receiving inspection, etc. This was partially confirmed by a subsequent I&E audit (reference 55). The general review is discussed in this section. More detail evaluation of TMI performance in the three areas above is discussed in sections III-C, D, and E.

The review included visits to the TMI site and the Met Ed corporate offices in Reading, Pa., discussions with a large percentage of the supervisors directly responsible for the TMI site, review of documents, depositions, procedures, LERs, I&E inspection reports, TMI audits, and I&E audits. The results of 2 days of interviews with key Met Ed personnel are contained in reference 67.

b. The TMI Organization

The corporate organizations at Reading, Pa., involved in the overview and assurance activities discussed in this paper are the vice president-manager generation, the manager-generation quality assurance (MGQA), the manager-generation engineering, the manager-generation operations, and the manager-generation maintenance. In addition to being responsible for the Three Mile Island plant, this organization is also responsible for operation of two fossil fuel power plants. An organizational change was made on March 5, 1979, to elevate the station superintendent position held by G. Miller to the position of manager generation station nuclear, reporting directly to the vice president, generation. All activities at TMI with the exception of the quality control group report to Miller. Terry Mackey, the superintendent of quality control at TMI, reports directly to Troffer, who was manager generation quality assurance.

With regard to the organization at TMI itself, the primary characteristics that should be noted here are that maintenance personnel are shared by both plants. There are common shift supervisors and separate technical support groups for each plant. As noted previously, the quality organization is dotted to the manager generation station nuclear and reports directly to Troffer.

The MGQA, in addition to having overall quality assurance responsibility, is also responsible for licensing, security, and training. The quality assurance staffs at Met Ed and TMI are not large. Troffer in his deposition (reference 59) indicated his licensing staff had 14 engineers and his quality assurance staff consists of six engineers and two administrative clerks. In addition, Troffer indicated he spends only about 25 percent of his time on quality assurance. Mackey indicated he had a staff of 14 including one clerk. The personnel on his staff are primarily inspectors who participate in the various quality control and inspection functions assigned at the site.

The NRC, as part of its FSAR review and in Section 17.1 of the Safety Evaluation Report (reference 36), approved the TMI organizational structure as meeting 10 CFR 50, Appendix B, requirements. Although our review generally agrees with the NRC assessment as to structure, it does not appear that there are sufficient engineering and inspection personnel assigned to the quality assurance organization to meet the intent of 10 CFR 50, Appendix B, or ANSI 18.7. For example, Dan Shovlin, superintendent of maintenance at TMI, indicated his normal maintenance crew was approximately 180 technicians (reference 67). In addition, he indicated another 24 to 30 Catalytic Construction Company technicians would normally be on-site doing maintenance and repair work. This number could go as high as 400 during a plant shutdown when fuel was being replaced. This meant an inspector crew of 12 to 14 persons had to monitor the activities of up to 500 to 600 people during active periods at the facility. It should be noted that prior to the accident at TMI-2, TMI-1 was just completing a fuel replacement sequence. As discussed elsewhere in this paper and also confirmed by Mackey and other personnel interviewed (reference 67), the quality assurance staff at TMI was quite often not able to perform its responsibilities of independent inspection and verification. The I&E report (reference 52) on the TMI-2 accident also confirms the lack of independent observation of surveillance activities, and it is considered this deficiency as a potential item of noncompliance.

Since some quality assurance-related functions are the responsibility of other organizations at Met Ed, some understanding of these organizations standing and activities is required. At the time of the accident, R. M. Klingaman was manager of generation engineering. In his interview with the Commission staff (reference 67), he indicated he had a staff of about 31 persons reporting to him who provided engineering support to all of the Met Ed generating stations. These personnel are based in Reading, Pa. This engineering staff does not have expertise in all areas, but can call on the GPUSC for assistance. He indicated that most of his nuclear engineering support comes from GPUSC. His office is responsible for maintenance of the change control system and assuring drawing changes are reflected in the as-built drawing. As discussed in section III-E regarding change control, and as indicated by the I&E audit conducted in late July 1979 (reference 55), a number of deficiencies exist in this system.

c. The Quality Assurance Plan

The quality assurance program for station operations for TMI-2 is described in Section 17.2 of the FSAR (reference 61). This operating quality assurance plan was approved by the NRC in its Safety Evaluation Report, Section 17.2, which simply indicates that the document meets 10 CFR 50, Appendix B, Regulatory Guide 1.33, and ANSI 18.7 requirements. The quality assurance plan in effect at TMI at the time of the accident was revision 7. The I&E audit of TMI in July 1979 (reference 55) reports that the changes in this document prior to the accident over that specified in the FSAR Section 17.2 had not been submitted to the NRC in the subsequent annual report as required and has cited Met Ed for an infraction. Discussions with the NRC Quality Assurance Branch indicated they felt that the plan was generally good, but they did have concerns about the implementation of the plan. Recent findings by the I&E audit, the Commission staff review, and the events that contributed to the TMI-2 accident confirms those concerns.

As discussed elsewhere in this paper, the quality assurance plan is only required to apply to hardware and activities considered to be "safety-related," and that is the extent to which Met Ed has applied the plan. For the TMI-1 facility the plan contains a specific list of hardware to be safety-related. In the case of TMI-2, the plan contains a summary of those systems or parts of systems that are included in the list and contains a seven-page description of the ground rules used in identifying which components are "safety-related." It also references a series of architect engineer documents and lists that specifically call out which pieces of equipment are considered in this category. Section III-B of this report discusses some of the problems associated with the application of the safety-related terminology and the confusion that appears to exist at the plant and within NRC as to which components in systems are to be covered in the plan.

The quality assurance plan addresses all 18 sections of 10 CFR 50, Appendix B, as required. Appendix B requirements do allow independent assessment functions to be conducted by other than quality assurance personnel. Met Ed has taken full advantage of this provision and has not taken advantage of the independent assessment capability of the quality organization. This has resulted in a number of weaknesses identified both by the recent NRC audit (reference 55) and by staff review (reference 64). Of particular interest is the documentation control system discussed in the NRC audit. Section V of the quality assurance plan entitled "Instructions, Procedures, and Drawings" assigns responsibility for maintenance of these document control systems to a number of Met Ed organizations with relatively little direct quality assurance involvement other than approval of procedures that the individual groups use to conduct their control. As shown by further discussion in this paper, the lack of independent (not in line) quality or safety advice to utility management to present unbiased evaluation of procedural changes, hardware changes, trend data, and nonconformances was a significant factor in the accident.

d. Met Ed Independent Review Groups

In addition to the quality assurance organization there are three independent review groups established by the TMI-1 and TMI-2 quality assurance plan, technical specifications, and corporate procedures. These groups provide overview of a number of activities as required by ANSI N18.7. They are the Plant Operations Review Committee, the Generation Review Committee, and the General Office Review Board.

The requirements composition and responsibilities of the PORC are delineated in Section 6.5.1 of the TMI-2 technical specifications (reference 66).

The PORC is the on-site review organization made up of members of the operating staff at TMI and include representation from operations, maintenance, radiation protection, and engineering. The organization does not include representation from quality assurance or any other independent organization not directly involved in the day-to-day activities at TMI. The PORC meets the basic requirements defined in Section 4.4 of the ANSI N18.7 specification for on-site review. The PORC conducts the initial review of the various documents, changes, procedures, LERs, etc., required by the NRC regulations. In particular, they review all procedures and procedure changes and appear to essentially perform the function of an engineering review board of a design and manufacturing organization. Their procedure review, as in other activities required by NRC, is limited to procedures listed in Section 6.8 of the technical specifications that are associated with the various safety-related activities of the plant. There is no indication that they participate in maintenance, operation, and repair procedures associated with non-safety-related equipment such as the condensate polisher or such vital systems as control drum drive mechanisms and iodine filters.

As discussed in section III-C, of this paper, a number of PORC meeting minutes were reviewed and indicate that the committee, during the period of 1978 and early 1979, was required to review and handle a large number of documents. The minutes of the meetings did not provide information regarding the depth of detail the committee went into in its review, but as discussed in section III-C, the large number of documents reviewed, particularly procedures in the time allocated, casts doubts as to the completeness of this review.

The staff, in interviews with TMI personnel, received two versions of what went on at the meeting which are somewhat conflicting. The superintendent of quality control, who only occasionally sits in on PORC meetings, expressed concern for the depth of detail that the committee went into in its review (reference 67). The chairman of the PORC for

TMI-2, Kunder, gave us a different picture indicating that the workload was considerably less and the content of the committee meeting minutes was quite complete. A review of PORC minutes indicates that the minutes are primarily a log of all the items reviewed, often a large number, and a basic statement that no unreviewed safety questions were found if that were the case. The minutes did not allow an evaluation of the depth to which the committee went to determine that unreviewed safety questions were not involved nor did they indicate what technical concerns might have been considered during the course of their discussions.

A number of incidents have occurred which question the depth to which the PORC goes and the true independence that it provides. (Some of these are discussed in sections III-C, D, and E of this report.) The situation of the EF-V-12A/B valves is probably the best example. In August 1978, the PORC reviewed a procedural change associated with these valves and did not recognize that the change which closed both valves at the same time put the operation outside of the technical specification.

This procedure was conducted many times between August 1978 and March 1979. At no time did the operators and supervisors realize they were outside of specification requirements. Since the PORC is the primary reviewer of procedures related to systems and activities under NRC cognizance, it appears that many of the problems and deficiencies discussed in this and other staff reports can be traced to an inadequate procedure review process. As discussed below, many of the other activities conducted by PORC including technical specification changes, LER reviews, configuration changes, etc., do get a more detailed review by other groups, but operating and surveillance procedures are primarily the responsibility of the PORC.

The requirements for the Generation Review Committee are contained in Section 6.5.2 of the technical specifications (reference 66). This group meets the basic responsibilities of an independent review group as required by Section 4.3 of ANSI N18.7/1976. The charter, responsibilities, and method of operation appear to duplicate the requirements spelled out in the ANSI document and were accepted by NRC as meeting the requirements for independent review. Met Ed procedure GPOO19, revision 1 (reference 68), provides the charter and assigns membership for this committee. There is no indication that the committee concerns itself with plant activities outside the scope of basic NRC requirements.

The committee is chaired by the manager-generation quality assurance and is co-chaired by the manager of generation engineering. Membership included expertise in the various disciplines required by NRC; however, a number of the people appeared to come from the audit organization within the quality assurance group of Reading, Pa. The committee is required to meet at least once every quarter during the initial year of operation; however, in fact, during the one-year period prior to the accident they appeared to meet every 2 to 4 weeks. The committee has four subcommittees that look at change modifications, technical specification changes, the quality assurance audit program, and procedural changes. In reviewing the minutes from the one-year period prior to the accident, it appears that although the GRC is to be an independent review of certain site activities, the members also utilize this mechanism to coordinate much of the activities that, are the responsibility of the Reading, Pa., staff, such as reviewing and approving major change modification packages that are submitted to Met Ed for approval.

It is difficult to determine from the minutes of these meetings and the minutes of the subcommittee meetings, the depth to which these committees conduct their review. As with PORC minutes, the GRC minutes tend to be listings of activities reviewed. There is some discussion on a few items, and there are attachments from subcommittee meetings that are more detailed and do provide some indication that at least for nonconformances, technical specifications changes, and LERs, there is some significant review. However, many times words are used that indicate the subcommittees reviewed the PORC minutes which, if that were the extent of the review, would probably be inadequate. Troffer in his deposition also indicated that the GRC does not generally visit the site as part of its review.

Because the committee does not appear to go into detail review of the procedures or operations <u>at the site</u> and only reviews safety-related areas, the GRC does not appear to be an adequate vehicle for management to assure they understand the overall health and operation of the plant.

The General Office Review Board was established in 1973 to provide an overall corporate review of the activities. Reference 69 defines the charter for this board and indicates the primary purpose of the GORB is to foresee potential significant nuclear and radiation safety problems and to recommend to the President how they may be avoided. Specific areas to be covered are proposed changes to the operating license, proposed facility design changes that involve unreviewed safety questions, technical specification violations, effectiveness of the PORC, and the adequacy of PORC determinations concerning unreviewed safety questions. The committee appears to meet quarterly and a review of the minutes indicate that the committee does meet the intent of the charter. Minutes were reviewed from meetings held on Dec. 20, 1977, through meeting No. 32 held on Jan. 10, 1979 (reference 65). The minutes indicate thorough detailed discussion is conducted on a number of items, but show that for PORC overview they primarily reviewed PORC minutes. At one point during the year, they did assign a member to sit in on PORC meetings to assure there was free exchange of information in the meetings and the activities were not dominated by any one individual.

On July 19 and 20, 1978, J. Thorpe, chairman of the GORB, and another member visited the TMI site and conducted this review. He reported there was a free exchange of information and there appeared to be competent personnel assigned to each PORC and that the GORB representatives were satisfied with PORC performance. A recommendation was made to continue this review on a quarterly basis. The GORB minutes also reflect that at various times the committee was concerned about staffing problems which apparently were nagging throughout the 1978 startup period. Although the GORB appears to be a useful vehicle for overview of nuclear safety problems, its relatively narrow scope does not provide Met Ed upper management an overview of the overall plant operations.

No quality assurance membership appears in the GORB charter nor appears to have presented documents before the board during the time period examined. Also, the board does not appear to overview, consult, or furnish expertise in such activities as system safety engineering to assure systematic safety analysis of the various activities under its cognizance.

e. TMI Performance

As shown above, TMI appears to have the basic organizations and structures to meet NRC requirements. In addition, the various Met Ed administrative procedures required by NRC to document how the plant is operated are generally complete and appear to define the operational processes adequately. Quality assurance is required to approve administrative procedures and appears to have done so.

However, when one looks at the detailed operation of the plant, there are many concerns that the overall activities were not being conducted well, and this poor performance may have been due to a lack of adequate overviews or independent assessment.

One example of this is in basic housekeeping, cleanliness control area. Section 5.2.10 of ANSI N18.7 requires that "housekeeping practices shall be utilized recognizing the requirements for the control of radiation zones and the control of work activities, conditions and environments that can affect the quality of important parts of the nuclear plant." Staff visits in May, July, and August 1979, all indicated a poor performance in this area. Of particular concern to the various staff members was the poor housekeeping noted in areas of the plant where radioactive contamination can be a problem. Visits by various staff members also noted basic industrial practice deficiencies associated with poor marking of piping and valves, valves in the reactor building exhibiting packing leaks, and ferrous components, valves, and piping generally covered with a layer of rust. These are all areas not covered by the quality organization and may indicate a lack of management overview of the operation or indicate that the total support provided the plant was not sufficient to meet its operating needs.

Another example is the staff analysis report on the condensate polisher where neither management nor the operators apparently understood the condition of a major facility system, its design, or how the system operated. More importantly, the supervision of the plant and the Met Ed management had no system in place to tell them that this was the case. The possible exception was the maintenance organization which does keep a good history of the various work requests that document the problems they have on non-safety-related equipment and which our review indicates did show that there was a long history of problems with this hardware. Unfortunately, there was no system to bring this history to management's attention.

Operational problems existed which are discussed in some detail in the NRC I&E report (reference 52) that are good examples of deficiencies that an independent assessment activity is designed to surface. The operators operated with the outlet temperature of the PORV above the procedural limit of 130° F since the fall of 1978, according to a discussion on page 1-1-5 of the I&E report. More than half of the portable radiation survey instruments were not operational when needed in the accident. As discussed in the previous section on I&E, there was evidence that this situation existed 9 months before the accident.

It is interesting to note that in the inspection report (reference 50) discussed previously under section III-B, the inspection results showing two of the four radiation monitors not working were not transmitted to a quality assurance representative at TMI nor was a quality assurance representative present at the exit interview, as the emergency preparedness and radiation safety activity is not an activity that is overviewed by the quality assurance organization. This is a excellent illustration of how the lack of independent overview of the entire plant prevents either NRC or Met Ed management from picking up, utilizing, and responding to signals that are forecasting potential safety problems.

In another example, the I&E report indicates that the shift foreman did not routinely review completed surveillance procedures to assure detail steps had been conducted. Lack of quality review of such procedures indicates, therefore, that there was essentially no independent review of this activity.

On Nov. 3, 1978, there was a complete shutdown of the plant, including reactor trip, caused by an individual turning off the electric power to the condensate polisher electrical panel when he thought he was turning on a light switch. Since this was not safety-related hardware and all emergency systems operated correctly, no LER was prepared and no significant I&E or TMI review of the incident occurred. The I&E report (reference 52) indicates that this might be an item for noncompliance since the steam generator went low in water level which should have resulted in a report to the NRC. Again, strict interpretation of regulations and requirements prevented this incident from being adequately reviewed so that management could fully understand what the incident was trying to say about the plant operation.

f. Audit

The results of two audits are available to tell us something about the overall operation of the Till plant. The NRC I&E conducted a quality assurance audit in late July and early August 1979 (reference 55). This audit shows a number of deficiences related to quality assurance areas and confirms a number of observations made by the commission's staff during its visits and review. Concerns, deficiencies, and infractions are noted in the report in a number of areas, including control of administrative procedures, drawing, change control, storage of material, procedure review, etc. In addition, the inspector looked at procurement control, storage, and handling which had not been reviewed previously by the Commission staff. Deficiencies were noted in both these areas.

The Met Ed quality organization and procedures include an internal audit program that is required in accordance with 10 CFR 50 Appendix B and ANSI/ASME N45.2.12-1977 (reference 60). The TMI-2 FSAR (17.2.18) states:

The Manager-Operational Quality Assurance is responsible for a system of planned and periodic audits to verify compliance with aspects of the Operating Quality Assurance Program. Audits are performed in accordance with written procedures and for check lists by qualified personnel who are independent of the area being audited. The results of these audits are formally documented and are reviewed by the Manager-Generation. This audit program also provides for follow-up on nonconforming and deficient areas found as a result of such audits.

The TMI audit program was reviewed by the Commission staff (reference 72), who found that it is organized and executed well. The audit program is documented in the TMI operational quality assurance program plan (reference 70), and in the generation procedures (reference 77). Thirty-one different areas are regularly scheduled for audits at intervals not exceeding 2-years. Audit schedules are issued every 6 months by the manager-generation quality assurance, from which monthly audit schedules are defined, audit procedures developed and approved, and audits conducted. A recent, Nov. 27, 1978, change to the procedures was made to ensure that audit reports are issued within 30 calendar days after completion of the audit.

GPU procedure 4015 contains requirements for audit team member qualifications and describes audit performance requirements. It requires audit plans, checklists, reports, and specifies the required content of each. Requirements exist for the performance of audits such as: entrance and post-audit meeting, use of checklist and information to be recorded. The responsibility for maintaining audit files and their content is specified.

Generation procedure 4015 describes the system used for closeout of findings. Each finding is assigned a due date during the post-audit meeting. Extensions up to 60 days are accepted/rejected by the lead audit engineer; extensions over 60 days require the concurrence of the supervisor-quality assurance. When response to an audit finding is one week overdue, a notice is sent to the cognizant party. (This was verified in a sample of audit findings examined by the Commission staff.) When 3 weeks overdue, another notice is sent, with a copy to the unit/station superintendent. When 5 weeks overdue, the manager-generation for quality assurance sends another notice, with copies to the generation and the unit/station superintendent. If this results in no action, the MGQA calls a meeting with the cognizant manager and party. Generation procedure 4015 also requires that an audit finding/recommendation status log be maintained and published monthly.

Six-month schedules for the past 2-1/2 years were available and up-to-date. A log of audits conducted since mid-1977 shows that all but a few audits have been conducted within their required frequencies, but that there was a trend of starting many audits a few weeks to a month late. There were 23 TMI nuclear power plant audits conducted in 1977, 22 in 1978, and 8 through June 1979. The accident disrupted the audit schedule.

Checklists were found to be comprehensive and should provide adequate guidance for performance of the audits. Reports were found to be well written and in compliance with requirements. The followup system was being used, although extensions were being granted too frequently. There were many cases where the auditor did not accept the proposed corrective action and held the finding open. As of June 5, 1979, 48 nuclear findings were still open; 6 since 1977.

Two of the TMI internal audit reports which were reviewed, audit report 78-25, Test Control (Nuclear), dated Dec. 15, 1978, and audit 78-23, Control of Special Processes, dated Jan. 15, 1979 (reference 71), demostrate the adequacy of the audit program, and also indicate that the surveillance test program is being conducted in accordance with regulation schedule requirements, although no assessment was made of quality controls' participation in test surveillance on the adequacy of the surveillance itself. Specific deficiencies in cleanliness procedures, quality control review of special operating procedures, and test result review in vendor purchase requests were noted.

Although audit reports and station information is widely distributed to Met Ed management, GORE, and GRC, no evidence was presented, nor was it evident that Met Ed management was acting on this information. The MGQA stated that he has seldom gotten involved in resolving findings. The vice president-generation had only been involved once or twice, at which time he issued verbal directions to the MGQA.

A cursory review of minutes of the General Operations Review Board audit subcommittee indicated that they did review audit status, but there was no evidence that indicated they were dissatisfied with the corrective actions being taken or that audit status was reviewed by GORE itself. The Generation Review Committee has a subcommittee to review audit status, but this amounts more or less to the audit organization reviewing its own work.

The findings of this Commission's staff relative to the audit activity were supported by the aforementioned I&E inspection of the TMI quality assurance program. The only item of noncompliance observed on the quality audit program was related to the lack of corrective action. This condition has existed over a long period of time as an internal Met Ed quality assurance effectiveness review dated Oct 20, 1977, (reference 72) indicates. "It (is) very evident that the timely resolution of audit findings is not being accomplished."

g. Equipment History and Plant Status

Reference 56 discusses a survey conducted by staff members of the Met Ed corrective maintenance history logs and other nonconformance and problem reporting systems to determine and examine the evidence of equipment failures over the past one to 2 years prior to the TMI accident. This investigation did demonstrate the difficulty of obtaining nonconformance data from the many nonconformance reporting systems that have been used over the past 2 years. This plurality of systems was in part a result of the change over from GPUSC to Met Ed responsibility, the difference in systems used to track reportable events, and the paper used just to accomplish work at TMI. An attempt was made in this study to gather data on all the major components or subsystems that appeared to contribute to actions that took place during the actual event. This review indicated that the maintenance department was attempting to accumulate data on various equipment problems; however, the data was not being used to collect trends to provide management with significant visibility on plant operation. It has only been since November 1978, that the data from work request documents, which are the primary day-to-day problem reporting form at TMI for equipment failures, has been collected on a computerized system that would allow easy examination of equipment failure trends. Recent discussions with the superintendent of maintenance at TMI indicate they have now initiated a hardware trend review program utilizing the data that is being accumulated in this system. In addition, previous data is being fed into the system from periods before November 1978.

There are a number of components that figured in the event that had been having problems prior to March 28, 1979. A rigorous problem-reporting, trend evaluation system might have prevented the accident or kept the operators from making erroneous conclusions. For instance, there were a number of work requests on the make-up pumps having to do with electrical problems. Also, work request CO 469, dated Dec. 12, 1978, indicates the pressurizer level transmitter indicator went high off-scale and that the instrument technicians worked on it, and it went bad again. Whether this affected the operator's mental thinking when he saw the pressurizer level go high could not be determined, but it was an indication of non-safety-related hardware that was having problems that in one way or another was involved in the event at TMI.

Reference 102 contains a long list of incomplete work items that apparently were still open following the turnover of TMI-2 from GPS to Met Ed. Not only was this list of open work not utilized by NRC to make its decisions regarding the status of the plant when it went commercial, but there was also no indication that the various TMI review committees overviewing the basic operation considered, or were concerned by, the amount of work yet to be done. This list, involving over $2 million worth of actual work, does represent a significant workload on the staff, engineering, operations, and quality, which must be considered when reviewing the overall activities in process at the plant at the time of the accident. Reference 73 includes a discussion of the staff review of this list and indicates that most of these items were probably open at the time of the incident. Many of the open work items date back to 1977. Twenty-nine items had been assigned to Met Ed for closure; another 90 had been assigned to Catalytic Construction. Included in the 90 were a number of items that had been written by the Occupational Safety and Health Administration (OSHA). In addition to these, there were another 494 "punch list" items covering a broad list of electrical, instrumentation, mechanical, structural, and piping items. Included in the list were a number of items concerning loose bolts on valve bonnets, bolts not torqued, items about poor or incomplete welds, some 90 weld histories not available, over 100 items of incomplete Engineering Change Modifications, and EDRs, and "as built" drawings not available.

h. Summary

The staff review of the TMI quality assurance, independent assessment, and overall operation indicates a number of deficiencies and weaknesses existed which contributed to the accident and the response of the operators and TMI management to that accident. The review indicates that TMI management did not apparently have a firm grasp on their operation nor did they utilize the systems of checks and balances available to them to obtain the necessary data to give them that understanding. Our review indicates that in the 6 months prior to the accident, there was an overwhelming amount of activity being conducted at the site that would have made it very difficult for the staff and organization in place to satisfactorily conduct and complete the tasks confronting it. In its review, the staff members conducting this assurance overview talked with many people at TMI and found that many were dedicated, knowledgeable engineers and technicians, who had legitimate concerns as to how well the activity had and was being conducted. It is quite evident that lack of Met Ed senior management overview and support of the overall TMI station operations was a contributor to the cause of the accident that took place on March 28, 1979.

Findings

- The Met Ed organizational structure, quality independent review groups meet basic NRC requirements.

- As implemented, the TMI independent assessment program involving quality assurance, and the review committees -- PORC, GRC, and GORB -- looked only at NRC required safety-related functions and, therefore, could not assure safe operation of the overall plant.

- Lack of quality assurance or other TMI independent assessment of non-safety-related hardware and procedures was a factor in the accident.

- Because of the limited purview of the review mechanisms, it is possible that Met Ed management was not fully cognizant of plant conditions and operations.

- Although the TMI internal audit program meets NRC requirements and is well done, Met Ed management did not assure that corrective action identified by the audits was initiated and completed in a timely fashion.

6. <u>The Application of "Safety-Related" in the Overview Process</u>

As can be seen from the previous discussion, the labeling of things as "safety" or "non-safety" has a significant effect on the way they are treated through the application of the quality assurance requirements. However, through this investigation it has been noted in conversations, interviews, and depositions, and in the review of documents, including the basic licensing documents (references 61, 36), and in many other working documents, that there is a range of different ways in which the terms safety, safety-related, safety grade, and systems important to safety, are used. It also appears that these terms, while generally applied for the same purpose are interpreted differently by different people in NRC and the industry and in some cases, even when applied to the same piece of hardware. To illustrate this situation, the following is quoted from reference 35 -- an internal NRC memorandum discussing a regulatory guide being proposed to correct the misunderstandings.

> The (regulatory) guide attempts to establish equivalency between the definitions of "important to safety" (see second sentence of first paragraph of the Introduction to 10 CFR 50 Appendix A) and "safety-related" (see third, fourth, and fifth sentences of the first paragraph of the 10 CFR 50 Introduction to Appendix B as applied to structures, systems, and components of interest to NRC that are included in nuclear power plants. While it may not have been the intent of the writers of these regulations to establish a difference in the meaning of these terms, users of these regulations, namely NRC reviewers and industry personnel, have perceived a difference and have based many decisions regarding the need and extent of quality assurance requirements in a nuclear power establishment on a list of specific SSC's (Structures, Systems & Components) (i.e., the Q-list) to which the provisions of Appendix B are applicable. At this point in time, we find it extremely difficult to see how NRC, through the mechanism of a Regulatory Guide with its inherently lower stature, can obviate these perceived differences in definitions without a corresponding **change in** the regulations. The proposed Regulatory Guide does not merely provide guidance on how to implement the regulations, which is its normal function, but rather attempts to modify the meaning of the regulations to be different than they have been perceived to be for several years.

As a result of the importance of the philosophy of "safety-related" to the overview process and overall plant safety, an investigation was conducted to determine whether there were misunderstandings in NRC and the utility regarding the philosophy, and whether a clear understanding existed as to what specific equipment was or was not safety-related. Several techniques were used as follows:

a. Interviews

Five persons in NRC were asked to define the terms safety-related, safety grade, and systems important to safety and to give examples to illustrate their 1efinitions. This sample included an assistant director, a licensing technical assistant, two branch chiefs in NRC, and an experienced licensing project manager.

Responses varied from concise statements to lengthy discussions indicating a broad range of misunderstanding. While the response dialogue was not helpful in defining these terms, it reinforces the observation that the terms are not clearly understood. Responses were as follows, with the respondents identified by numbers and not in the order listed above.

A. Safety-related:

1. Those systems required to prevent or mitigate an accident.

2. Equipment we have given credit to mitigate an accident; equipment used to respond to an accident; infers quality of equipment.

B. Safety-grade:

1. Refers to all the requirements for all safety-related systems; includes seismic qualifications, environmental qualification and all quality assurance requirements.

2. Same as A-2 above and designed for seismic considerations such as Safety Shutdown Earthquake (SSE); must meet single failure criteria as defined in general design criteria 10 CFR 50, Appendix A; has to meet IEEE 279,308 and standards required by 10 CFR 50.

3. Term is used in NRR but not used in standards or regulations. Term slipped into draft guide on Residual Heat Removal (RHR), but will not be used in final guide.

4. Implies application of seismic I classification and quality classification to specific items of equipment identifed as safety-related.

C. Systems important to safety:

1. Have a bearing on safety, but are not necessarily required to prevent or mitigate accidents.

2. Important piece of equipment relied upon to perform some safety function.

Examples given by the same respondents as above:

A. Safety-related:

1. Emergency core cooling system (ECCS), class lE electrical systems (defined by IEEE-308; however, regulation (NRR) uses the term class lE to describe all electrical, instrumentation and control systems that are safety-related). Also there are some safety-related electrical systems that are not considered to be class lE. Off-site power, required by GDC17 (Appendix A to 10 CFR 50), is not class lE, but it is safety-related (off-site power is also not classified as seismic category I).

2. ECCS, pressurizer safety valves, RHR, and systems identified in the Regulations. (When specifically asked about the treatment afforded auxiliary feedwater systems (AFS), which are not

explicitly identified in the regulations, the response was that this is a come-lately; "We currently say that AFS is safety-related and must be safety grade.")

No examples were cited to illustrate the term "safety grade." No one could reference a document in which these terms have been defined or explained.

Two of the persons who played major roles in the writing of one of the appendices referenced in this investigation were contacted for definition of this terminology. Both declined to provide definitions.

In addition to the structured interviews, the interview with Terry Mackey, supervisor of quality control at TMI, produced the following information quoted from reference 67.

DWIGHT REILLY: Terry, would you give us your thoughts relative to this problem of identifying part of the plant as safety-related and part of the plant non safety-related and how this affects the overall safety and really the reliability of the plant?

TERRY MACKEY: Well, the origin of safety related versus non-safety-related systems in both Unit 1 and Unit 2 at TMI and I believe throughout the industry is the architect-engineer's responsibility. He defines what systems are safety-related and what are not and to what extent. If you'll review GP1008 (TMI QA Plan) you will see Unit 1 has specific components listed in about 30 or 40 pages. Just what components are nuclear safety-related and any work therefore or any changes in the operation of these components constitutes a change to a safety-related system. In Unit 2, which I am most familiar with, Burns and Roe published their specification SP 88, which designated what systems and components were to be considered nuclear safety-related and they there have a break-down of yes they are nuclear safety-related, yes, and, yes it's caused much confusion and conflict here because the yes and means it's within scope or without QC scope, within the QA function only to the extent that it is mounted for seismic events-.

b. Depositions

A number of depositions were examined to obtain an impression as to how those key people used and thought of terms such as safety-related and safety grade.

A comment from reference 16 is presented below as an example of the understanding of safety-related.

QUESTION: What are the requirements or characteristics of a safety-related device besides redundancy?

ANSWER: It varies because safety grade is not a well-defined term. You mentioned one, redundancy. It almost always included what is known as seismic class 1 which means it is designed to withstand the design earthquake for that facility.

Another comment, from reference 19, gives still another example of the understanding of safety-related.

> <u>ANSWER:</u> I don't have an exact definition of safety-related that I could give you. I could say -- I will make an attempt.
>
> <u>QUESTION:</u> Please do.
>
> <u>ANSWER:</u> A safety-related component would be one that was necessary to mitigate the consequences of a transient or accident, that is, to prevent violation of safety limits. Or would be relied upon to prevent the release of radioactive material.
>
> <u>QUESTION:</u> Let me read from the first paragraph of your resume and ask you if this expresses the same definition or another definition of safety-related. The reactor systems branch is responsible for evaluating the capability of reactor safety systems needed for safety shutdown during normal and accident conditions, including the performance of emergency core systems.
>
> <u>ANSWER:</u> Yes. Safe shutdown would be another aspect of what I was saying.

Also, during some depositions specific questions were asked about the existence of a list of TMI-2 hardware that would identify those items that were safety-related. When the existence was acknowledged, a specific request was made for a copy of the list, as in depositions (listed as references 16, 9, for example). In not a single case has such a list been produced. The closest thing has been the receipt of pages from section 17 of the TMI-2 FSAR which simply lists table 17.24, "Summary of Quality Assurance System or Partial System List," and procedure GP1008, the "Operational Quality Assurance Plan Appendix D," which contained seven pages of descriptive material defining ground rules for determining if equipment is safety-related (reference 70).

c. Analysis

The way that equipment is classified governs the way it is considered by its owners; it determines how well it is designed, how well it is analyzed, how it is handled, how related procedures are generated, how people are trained to use and care for it, and how it is treated when it has problems. The classification also effects the way NRC treats it in the licensing process, inspections, and in problem reports.

When there is a problem in understanding the classification and a lack of clear understanding of how the equipment is classified, there is potential for errors, omissions, and mistakes which could lead to accidents that would adversely affect the health and safety of the workers and the public. One example of how this problem of understanding can cause serious problems is highlighted by the TMI-2 accident. Post accident investigation has revealed that the PORV was most often classified as non-safety-related hardware (reference 26). Had it been classified as safety-related, as is now believed to be its true status, its design and application would have been more carefully reviewed and most significantly, its problems would have been reported more completely which could have led to more attention to its hardware-related problems, to related operational and emergency procedures, and, perhaps, to recognition in the training program.

Findings

- Significant misunderstanding exists among NRC and TMI-2 personnel regarding the meaning and application of terms such as "safety-related," "safety grade," and similar terms.

- Misunderstanding exists among NRC and TMI management and project personnel as to what specific hardware is considered safety-related and what specific document defines that hardware.

- The lack of clear designation of "safety-related" equipment and specifically what that means contributed to inadequate hardware and procedure review, and failure analysis and corrective action that are necessary to assure safe operation of the plant.

C. PROCEDURE SYSTEMS, GENERATION, AND EXECUTION

1. Background

10 CFR 50, Appendix B, requires that procedures be prepared and used for activities affecting the quality of safety-related structures, systems, and components. NRC Regulatory Guide 1.33 (reference 10) provides guidance concerning minimum procedural coverage for plant operating activities, including related maintenance activities and contains a listing of typical procedures. Regulatory Guide 1.33 also endorses ANSI N18.7-1976 which is directed primarily toward administrative controls and quality assurance associated with safety-related activities, equipment, and procedures. ANSI N18.7 provides specific requirements concerning preparation review, approval and control of procedures as well as procedure content. The requirements and guidelines for preparation, review, approval, distribution, and revision of procedure at Till is contained in station administrative procedure 1001, "Document Control," revision 18, dated February 21, 1979 (reference 20). This procedure also contains requirements concerning responsibility for procedure performance.

Operating procedures address such functions as heatup, reactor startup, shutdown, auxiliary plant functions, electrical lineup, emergency procedures, etc. Surveillance procedures are used in the periodic testing of plant equipment to assure its operational status. According to personnel interviewed at TMI-l, the procedures for TMI-2 were developed based on TMI-1 experience. The TMI-1 procedures were given to Babcock & Wilcox with TMI operator comment. B&W provided a set of draft procedures for TMI-2 using this input. A test working group was established to take these procedures and put them into final form suitable for step-by-step operator use. The set was then reviewed by the PORC and approved by the unit superintendent. No detailed evaluation was made of the process to develop the initial set of procedures, but the role of PORC and other groups involved in the review of procedures was evaluated since January 1978.

2. <u>I&E Review</u>

The NRC Office of Inspection and Enforcement is responsible for the review of plant procedures. Inspection procedure 42700B, dated Jan. 1, 1979, contained in the I&E manual (reference 40), describes the methods used by NRC in their review. This inspection procedure is only required to be used once a year and the frequency may be reduced. This procedure requires that samples of all types of plant procedures be reviewed, but requires that only two be reviewed for technical content. This minimum review of procedures was verified by Haverkamp, NRC inspector for TMI (reference 17).

<u>QUESTION:</u> Are you fairly familiar with the licensee's procedures?

<u>ANSWER:</u> I am not that familiar with the operating procedure only a selection of them, on an infrequent basis, just to review the procedures, but my previous inspections did not require that I look at each procedure and be familiar with the details of those procedures.

<u>QUESTION:</u> Is your basic responsibility then to determine that the licensee has procedures and then selectively determine what the procedures are and whether or not the licensee is complying with them?

<u>ANSWER:</u> That is correct, we do, other inspectors besides myself, review for example, maybe 10 operating procedures for technical adequacy, maybe five, a certain small number of operating procedures about once a year and do a technical review.

<u>QUESTION:</u> What percent of the procedures are we talking about when we talk about 5 or 10 procedures?

<u>ANSWER:</u> I would say less than 5 percent, on the average of 1 or 2 percent of procedures.

Format and technical control of facility procedure was one of four elements of a NRC inspection of TMI-1 and -2 conducted May 30 through June 2, 1978. The inspection involved 22 inspector-hours on-site by one NRC inspector. This report indicated that one TMI-2 operating procedure, one TMI-2 emergency procedure, and five TMI-2 alarm procedures were reviewed for technical content. No nonconformances were noted for this area (reference 75). As discussed previously, procedures are not reviewed by DPM or DSS as part of their license review, and although no review of specific I&E review of procedures prior to licensing was made, the I&E manual indicates the procedure review requirements during the prelicensing period are similar. To that discussed above, it appears, therefore, that no effective NRC review is conducted.

3. TMI Preparation and Review

The TMI procedure for the preparation, review, approval, and distribution of procedures, (administrative procedure 1001) appears to be adequate for the intended purpose, except it requires very little if any independent review. It does require that procedures that are safety-related, involve radiation exposure to personnel, or involve potential or actual release of radioactivity will be reviewed by the Plant Operations Review Committee and approved by the unit superintendent. Some procedures that are on an environmental impact list are reviewed and approved by the manager-generation quality assurance. Since the PORC is composed of representatives from the operations and technical support organizations (the same organizations that prepare procedures and procedure changes) they can hardly be considered an independent review element (reference 66, 67). This lack of independent review has resulted in procedures and changes thereto being released without adequate review.

The minutes of three TMI-2 PORC meetings (reference 74) were reviewed to determine the scope and depth of the reviews performed by PORC. The meeting minutes reviewed were:

- Minutes #256 (Feb. 20, 21, 23, 24, 1978)
- Minutes #78095 (Jan. 29, 30 and Feb. 1, 2, 1979)
- Minutes #7922 (May 28, 29, 30, 31, and June 1, 1979)

Item one discussed eight meetings held during the period covering a total of 8-1/2 hours. The report basically states that 7 work requests with their procedures, 41 procedure change requests (PCR) and temporary change notices (TCNs), 2 special operating procedures (SOPs), 15 test procedures, 6 plant mods, and 5 reportable occurrences were reviewed. No detail is given regarding the discussions held although copies of pertinent LERs and technical specification change justification documents are attached. Item two covers a series of seven meetings totaling 7-1/2 hours. This series of meetings reviewed a total of 58 procedures or procedure changes plus other documents. As in the previous item no detail is given regarding the depth of discussion or concerns generated. Item three documents seven meetings covering a total of 4-1/2 hours. Seven special operating procedures applicable to the cleanup process plus other documents were reviewed. It appears from this review that on the basis of volume alone, PORC only gave cursory review to items, particularly procedures.

In addition to the PORC review, procedures and changes thereto are reviewed by a subcommittee of the TMI-2 Generation Review Committee. The scope of their review only includes safety-related matter (reference 68). The committee does review and concur in nuclear safety evaluations and environmental impact evaluations associated with safety-related procedure but does not "get too much into the details of the procedure" (reference 67). Operating procedures and changes thereto that are not safety-related such as operating procedure 2106-2.2, "Condensate Polishing System," revision 9, dated March 21, 1979 (reference 78), are only reviewed and approved by the unit supervisor. No independent review is required (reference 20). Weaknesses in procedure 2106-2.2 are discussed in reference 79, the technical staff analysis report on the "Condensate Polishing System."

To determine to what extent defective procedures contribute to events that require preparation of licensee event reports, all LERs from four nuclear power plant units were reviewed. The results of this study show that approximately 11 percent of the reportable events occurred because of defective procedures. The plants and percent of LERs charged to defective procedures are Davis-Besse 12 percent, Rancho Seco 10 percent, TMI-1 10 percent, and TMI-2 11.5 percent (reference 73).

One example of inadequate review is procedure change request 2-78-707. This change to surveillance procedure 2303-M27A/B resulted in both emergency feedwater block valves (EF-V-12A/B) being closed at the same time. The closure of these valve was in violation of the TMI-2 technical specification (reference 80). A more vigorous change system that included independent review by individuals not responsible for the procedure preparation and operation, such as quality assurance, could have possibly detected this shortcoming.

Other examples of inadequate procedures are found in LERs (reference 81). LER 79-07/3L concerns inoperable traveling water screens due to significant buildup of debris because the procedure did not require one screen to be continuously operable during periods when large amounts of debris are present in the river. LER 79-09/3L concerns not performing a surveillance procedure per technical specification. This occured due to lack of clarity in the shutdown procedure, in that it did not adequately "call out" the performance of this event-related surveillance.

4. <u>Quality Assurance Overview of Surveillance Testing</u>

TMI administrative procedure 1001 does state in part that "the Quality Assurance Department has the option to surveil any and all procedures. Procedures chosen for quality assurance surveillance will be indicated with the words 'QC hold points indicated' or 'Performance to be observed by Quality Control. Notify QC at least four hours prior to starting task' on the cover sheet."

During the interview with Terry Mackey, TMI-2 quality control supervisor, it was learned that there was very little quality control participation in the surveillance activity (reference 67). ". . . our involvement in the actual performance of the surveillance procedures is limited to a random, well not quite random, selection of specific procedures . . . ;" ". . . that is done sporadically whenever I feel like I've got personnel available and we have not looked at a surveillance procedure being performed and we would just go watch and see that the procedure was indeed followed in the performance of the test."

LER 79-01/IT dated Feb. 14, 1979, concerns an operator not reviewing surveillance results versus acceptance criteria. TMI-2 technical specification, paragraph 3.1.2.9, requires that the boron concentration in the boric acid mix tank not be above 13,125 ppm. The data sheet shows that the concentration was found to be 13,788 ppm and the data sheet was accepted with no action being taken. It is for the purpose of preventing such errors as this that many industries implement a quality control system to assure that operator actions are reviewed. NRC, through the endorsement of ANSI N18.7/1976, endorses the policy of allowing second-line supervisory personnel to perform the quality control function of verifying that operator actions are in compliance with requirements. At TMI, first-line supervisors are allowed to perform this function (reference 20). This review function is inadequate in another respect since the supervisor/shift foreman does not routinely review completed surveillance procedures except for the completed data sheets. Surveillance procedures contain steps which, if not specifically completed and verifed to be satisfactorily completed, could leave an engineered safety feature system in an inoperable condition (reference 52).

5. Quality Assurance Overview of Maintenance and Repair Procedures

NRC Regulatory Guide 1.33 (reference 10) requires that "maintenance that can effect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances." TMI administrative procedures require and the staff review indicates that quality control is generally involved in the review on maintenance and repair procedures on safety-related hardware. Met Ed had chosen not to use quality control on other hardware, a policy that resulted in maintenance on many critical plant systems not being covered by quality control. A review of maintenance data indicates these systems include air handling, condensate, circulating water, feedwater, health physics, hydrogen recombiner, control rod drive mechanism, station service air (reference 82).

All maintenance and repair work at TMI is controlled by their work request system, as described in TMI administrative procedure 1026, revision 9, dated May 17, 1979 (reference 83). A procedure is prepared for each work request. The work request and procedure is reviewed by quality control and quality hold or inspection points added only if it involves a safety-related item. An example of a significant maintenance procedure not requiring quality control verification is TMI-2 maintenance procedure 2401-5.2 dated March 11, 1977, Control Rod Drive Mechanism Removal and Replacement (reference 84). The drive is an electromechanical device consisting of an electrically driven roller nut assembly (rotor) inside a pressure vessel, a stator, a translating leadscrew, and an external position indicating system. This procedure contains many detailed steps and many caution notes indicating that all operations must be very carefully performed. There are several requirements to torque bolts that should be verified. Other significant operations such as installation of 0-rings, lock wiring, and precise positioning of heavy equipment are required. All this is to be done in a hazardous area. It is hard to believe that an operation as difficult and as significant as this would be performed with no independent verification.

During the staff interview with Terry Mackey, TMI quality control supervisor at TMI on July 6, 1979 (reference 67), he indicated that he was unable to cover all maintenence and repair procedures requiring quality control verification. He mentioned a recent case in which a contractor was brought to perform nondestructive testing of pipe welds to American Society of Mechanical Engineers (ASME) code requirements. He stated that he was not able to provide continual surveillance of the inspection operation and that the data later was found to be in error. Mackey also stated in this referenced interview that in many cases quality control personnel are not available when called to cover hold points in maintenance operations. They attempt to followup at some time during the operation, and do review the completed work request to assure that all data is complete and all steps were conducted (reference 67). TMI quality control personnel only work the day shift although maintenance and surveillance procedures are sometimes conducted on the second shift. A quality control representative is on call but reviews are usually conducted the next day. Also, standing maintenance operations such as repacking of valves, which are performed quite often, are usually verified by quality control after the fact (reference 67).

Findings

- There is essentially no NRR review of detailed utility procedures. Reviews are limited to assuring that a proper list of procedures is available and a utility procedure review system is in place.

- I&E review of procedures is limited by intent to about 5 percent of operating and emergency procedures, and changes to procedures identified by the utility as impacting the technical specifications.

- The PORC is the primary procedure review organization. Current PORC membership and review practices appear to preclude adequate independent review of procedures associated with safety-related systems.

- Lack of TMI quality control overview of the preparation and conduct of surveillance procedures can preclude detection of omissions, mistakes, and unsafe practices by the utility.

- A small utility quality control staff precludes adequate verification (inspection) of maintenance and repair of safety-related systems and components.

- There is no independent review of verification of maintenance and repair procedures involving systems not identified as safety-related, but which may be important to safe and reliable plant operations.

D. NONCONFORMANCE REPORTING SYSTEM

1. Requirements and Regulations

10 CFR 21 (reference 85) provides the primary requirements for reporting abnormal occurrences, problems, or failures that occur relative to a nuclear power plant. This part of the code applies to all activities licensed by NRC and requires reporting if any:

> facility or activity fails to comply with the Atomic Energy Act of 1954 as amended or any applicable rule, regulation, order, or license of the Commission relating to substantial safety hazards or that the facility, activity or basic component supplied to such facility or activity contains defects which could create a substantial safety hazard.

Paragraph 21.3 defines basic component for a nuclear power reactor to be a:

> plant structure, system component or part thereof necessary to assure: (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shut down condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to those referred to in paragraph 100.11 of this chapter.

The document further defines "substantial safety hazard" to be:

> a loss of safety function to the extent that there is a major reduction in the degree of protection provided to the public health and safety.

These definitions, when coupled with the definitions of 10 CFR 50, Appendix B, have tied the reporting system to "safety-related" hardware and apparently helped narrow the definition of "safety-related." 10 CFR 50, Appendix B, also discusses nonconformances and corrective action, but merely states that the utility shall have a system to control review and correct nonconforming items.

NRC Regulatory Guide 1.16 (reference 97) provides further guidance on reportable events and redefines "abnormal occurrence" as any "reportable occurrences that are determined by the Commission to be significant from the standpoint of public health or safety." This guide provides the instructions and requirements for submittal of reportable events in a licensee event report form. It further delineates those types of events, failures, and accidents that must be reported in 24 hours by telephone, followed up with written confirmation the following work day and a written report in 2 weeks. These include significant failures or operations that are outside of technical specification limits or failures that hinder safe shutdown, and events like civil disturbances, etc. Less significant reportable events may be submitted in a 30-day written report although the basic format and content of the

LER are utilized. Regulatory Guide 1.16 also provides instructions as to what should be contained in startup reports, monthly and annual operating reports, and other routine reports.

The result of these requirements is that the utility primarily reports events or failures associated with safety-related hardware or which put operations outside of the technical specification rules under which the utility is operating. In addition, the distinction between items requiring 24-hour submittal or 30-day submittal means that even within the "safety-related/technical specification" designation, there are more critical and less critical events and, as will be shown later, the NRC response emphasis is on the 24-hour reportable event. The requirements, therefore, narrow significantly the NRC overview of failures or conditions that exist at the plant and, as has been discussed previously, prevent critical review of plant systems that are important to its safe operation. Systems like the condensate polisher that can initiate a major failure are ignored as are failures to critical valves such as the PORV unless such failures result in some other condition that requires a report.

2. <u>TMI Preparation and Review of Reportable Events</u>

The responsibilities and procedures for identifying and processing reportable nonconformances or events at TMI are defined in the TMI operational quality assurance plan and in Met Ed generation procedures GP 0029 and GP 0075 (references 86, 87). In addition, section 6 of the TMI-2 technical specifications defines the responsibilities for review of these events by the PORC and GRC. As discussed in section III B-5, the General Office Review Board does sample or audit the LER activity.

GP 0075 assigns responsibility to the manager-generation quality assurance for assuring that "GP 0029 is implemented such that deviations are identified, evaluated and reported to the NRC if they involve a defect or noncompliance per 10 CFR 21.3." GP 0075 indicates deviations and noncompliance may be discovered in many ways by many people and should be reported per GP 0029. The procedure indicates the MGQA is responsible for assuring a deviation is evaluated by generation engineering or another appropriate division, although it need not be evaluated if already "adequately identified to NRC." Results of evaluations are to be reported to the licensing section in quality assurance that essentially tracks and prepares LER transmittal documentation, unless the evaluation is conducted by it. The MGQA is also responsible for informing the vice president-generation when it is determined an event or defect is reportable.

GP 0029 defines the procedure to "ensure prompt and effective action on all nonconformance items under the responsibility of the Generation Division." This is a very simple document that essentially says each manager and station/unit superintendant is responsible for correcting and documenting any nonconformance identified as within the scope of his responsibilities. It provides a table that shows a number of ways that a nonconformance or defect can be reported and gives the <u>immediate supervisor/foreman</u> the responsibility to decide if the nonconformance is potentially reportable under 10 CFR 21. A simple questionnaire is provided to guide the supervisor/foreman in his decision-making. This questionnaire simply answers the three requirements out of 10 CFR 21.3 quoted previously, plus a question as to technical specification applicability. GP 0029 also says quality assurance shall assure an NRC noncomformance report is written and corrective action is taken per GP 4012 (reference 88). A review of GP 4012 shows that the NCR form is primarily intended for use by quality assurance and quality control, but anyone may use it. Under responsibilities, it gives the MGQA responsibility for resolution of nonconformances within quality assurance and the superintendant-quality control responsibility for ensuring nonconformances identified by nonquality control, on-site personnel are resolved in a satisfactory and timely manner. The document also gives quality assurance certain stop work authority.

These documents, discussions with TMI personnel (reference 67), review of equipment histories, etc., indicate the identification of a failure, defect, or nonconformance that requires reporting is a very loose, nonrigorous system. During startup, a number of different documents were used to identify a problem. The current TMI-2 work control system uses an International Business Machine (IBM) work request form that serves both as a report of equipment malfunction plus a mechanism to repair or correct the problem. Since the generation procedures stress that NCRs are quality control documents, and our discussions confirm that the NCR's form is used primarily by quality control, then other documents, work requests, memorandum from operators, etc., are needed to trigger an LER. There appears to be no one document at the working level that says something failed, a limit was exceeded, an operation was not conducted, etc. It is particularly unclear how operations out of technical specifications limits are recorded since quality assurance is not part of operations.

Once a potential reportable event is surfaced, by whatever means, section 6 of the technical specification (reference 66) defines the independent review required. Section 6.5.1.6 requires the PORC investigate all violations of the technical specifications and review events requiring 24-hour notification to NRC. No mention is made of the review of 30-day LERs.

Section 6.5.2.8 requires that the GRC review violations of regulations, codes, and technical specifications having nuclear safety significance, significant operating abnormalities that affect nuclear safety, and events requiring 24-hour NRC notification. They are also responsible for reviewing "all recognized indications of an unanticipated deficiency in some aspect of design or operation. . . that could affect nuclear safety."

A large number of PORC and the 1978 GRC minutes were reviewed. They indicate the PORC and GRC do review LERs and the GRC also reviews a number of NCRs. The minutes lack sufficient detail to determine the depth of review given, although it does appear they primarily review the basic LER as written.

A review of the 23 1978 GRC-2 meeting minutes plus a number of special, limited agenda meetings shows that in the GRC:

- An LER may be listed with an action to obtain additional information on cause/corrective action.

- An LER may be approved for submittal to NRC. This implies that GRC reviews/approves each LER before submittal to NRC. A check of LER submittal dates vs. GRC meeting dates on a given LER show this is not always true.

- An LER may be determined to not be reportable to NRC and the LER number assigned to another event.

- An LER may be closed by GRC-2 in that no safety concerns are found to exist.

- One LER was found where GRC-2 had a generic concern and requested further information on corrective actions.

In the PORC minutes, the LERs are generally listed and attached. No mention is made of reviewing detail failure analysis, therefore, the analysis and corrective action given in the LER appears to be the prime source of data. As discussed below, the content of LER does not itself support adequate review. There is evidence that the Nonconformance Subcommittee of the GRC does track NCRs, and a few subcommittee reports are available that indicate quality assurance is tracking and closing these. None of the documents provide information that the utility conducts a rigorous failure analysis and corrective action program to assure all aspects of what caused the event or failure and what total corrective action is needed are identified.

As discussed previously, the GORE provides an independent corporate audit function to foresee significant safety problems. GORB meeting minutes for meetings 29, 30, 31, and 32 (reference 65) were used to assess how well GORB supports the failure reporting system for LERs. This review found:

- PORC minutes (including entries regarding LERs) are reviewed by GORE secretary and summarized for GORE members in a report.

- The PORC chairman makes a summary presentation of PORC activities including questions of reportability of incidents.

- GORB reviews LERs as to whether or not there are any unreviewed safety questions.

- In these meetings, GORE concurred with action taken and found no unreviewed safety questions on 76 LERs. GORB minutes show additional questions were asked on 8 LERs. These questions were on LER details or procedures and were answered in the meetings.

GORB's function on LERs apparently is a check and balance of how safety questions and reportability have been addressed by PORC. GORB's review of LERs is brief and shows little or no concern for failure analysis and corrective action on individual LERs.

3. <u>LER Content and Format</u>

In 1973 a **computer-based data** file of informa' = 'n was established under AEC to provide a central data source for off-normurl events in the nuclear industry. This data file reflects the reporting requirements of 10 CFR 21 and NRC Regulatory Guide 1.16. Detail requirements for completing the LER form, which is the heart of the system, are covered in NRC NUREG-0161 (reference 89). The LER format and content is limited to items that are safety related. This means a utility is obligated to report a safety-related event on an LER, but it is not required to address nonsafety-related items, however significant, in the LER. This can and does lead to omission of significant technical data from the submitted LER form which in the final analysis impacts plant safety and the ability to recover from an accident. The LER for the incident at Davis Besse-1 in September 1977 addressed a half trip of the steam and feedwater reactor coolant system (RCS) and the cause was attributed to an electrical control problem. During the event, the PORV valves opened as required, but failed to close as required. The LER mentions the fact about the PORV valve, but the LER (correctly under NUREG-0161) did not include any failure analysis or corrective action relative to the PORV since it is <u>not safety-related.</u> The LER was correctly coded by Davis-Besse under system description as "I&E" -- other instrument systems required for safety. Another example, also a Davis-Besse LER (reference 90), illustrates even more graphically how LER format and content inhibit the exchange of vital technical information. This LER involved a reactor/turbine trip resulting from a procedure error which called for tripping of the turbine generator output circuit breaker. The LER does not mention the fact that loss of pressurizer level off the low end of the scale occurred during the event. Pressurizer level indication played a key role on the March 28, 1979, accident at TMI-2 and is, indeed, an essential parameter to understand in the safe operation of a B&W plant. The most interesting point is that operations/training personnel at TMI-2 had reviewed this LER before March 28, 1979. TMI-2 people factored the information about the circuit breaker into their operator training program, but could do nothing about the loss of pressurizer level indication since the LER was <u>devoid</u> of this significant information (reference 67).

Another deficiency of the LER format and content is the failure to enforce a rigorous requirement for followup on LERs where the initial information is incomplete or corrective active/failure analysis is still open. Page 39 of NUREG-0161 states that when LER information is i ncomplete, the initial LER should be labeled "Interview Report" and should indicate when an updated LER will be submitted. However, there is no data block on the LER requiring an interview report entry or a data block for a followup date. Item 17 of the LER includes an LER revision number but does not indicate if a revision number is required. A review from an LER computer printout (reference 91) for all 72 TMI-2 LERs through Feb. 17, 1979, shows that existing LER format/content requirements for followup data are not effective. Only one revised (-1) LER is included while all others are the initial issue (-0). Of the other 71 LERs, a reading of the text cause descriptions indicates that at least 19 LERs should have had followup LERs to close out the failure analysis/corrective action. A specific example is TMI LER 78-016/IT-0 (reference 92) where the -0 indicates the initial issue of the LER.

This LER involved the loss of two independent river water loops resulting from a "sneak" current path when a light bulb burned out. The corrective action was to replace the light bulb and "nuclear river water pump circuit designs will be reviewed." This obviously required a followup LER but there is no evidence that such an LER was ever prepared. This LER format/content deficiency is a definite contributor to poor quality failure analysis/corrective action by the utilities and NRC because vital information is being excluded from the system.

Another deficiency of the LER format/content from the standpoint of NUREG-0161 is the LER form itself (exhibit A of 0161). The form requires an entry of the name of preparer and phone number, but the entry is not made as a data block. Thus whether the information is in the computer or not, the available computer output (reference 91) shows that this information is not displayed. The LER computer outputs are the principal channel for exchanging failure data between the utilities. The absence of name of preparer and phone number places the user in the position of being able to use <u>only</u> what appears in the LER computer printout or possibly getting firsthand information after a time-consuming effort to find out who wrote the LER.

A detailed review of eight TMI-2 LERs (reference 93) chosen at random for compliance with NUREG-0161 was made to assess how well the form was being implemented.

<u>LER DEFICIENCY</u>	<u>NUREG-0161 REQUIREMENT</u>
No call-out of procedure/ procedure number under Description (item 10).	Required by paragraph A.2, page 11 Event.
No attachment included giving detailed description of the event (reference item 10).	Required by NUREG-0161 procedures, page 3.
Attachment not clearly identified.	Page 5 requires attachment be identified by LER number, licensee name, facility name, and docket number.
LER not identified as "Interview Report" when cause and corrective actions incomplete (item 27).	Note 1, page 39, requires "Interview Report" description and date for expected submittal of necessary information.
No description of measures to assure that similar components at all plants -- TMI-1 and TMI-2 -- at reporting facility are acceptable (item 27).	Required by item D, page 38.

Cause code (item 12) incorrect. Code was "component failure" "defective procedure" (reference LER 78-014/3L).	Do not use "component failure" when cause can reasonably be attributed to installation errors (paragraph E, page 15).

The above described deficiences were fairly common for the LERs examined and indicate considerable room for improvement in using the existing LER reporting system. The responsibility for correcting these deficiencies must rest with Met Ed and NRC; Met Ed to follow LER instructions correctly and NRC to assure compliance with approved requirements.

4. TMI Review and Control of Nonreportable Problems and Failures

As discussed previously, TMI has had a number of different documents to report problems and failures. During startup, GPU and B&R problem reports, field questionnaires, etc., were being used by GPUSC while the basic NRC work request system was used by Met Ed. Reference 56 describes an attempt by the staff to review the problem history of equipment that was a factor in the accident. Satisfactory data could not be readily obtained as the data was in many forms, in many places.

The staff did find that the Met Ed maintenance personnel were keeping good track of work done on equipment through their work request system. This system is described in station correction maintenance procedure 1407-1 (reference 94). It is a well-ordered system and since November 1978, the work request data has been computerized and IBM printouts of "corrective maintenance component history reports" have been available. There is no indication and TMI people confirm that neither the PORC, GRC, nor GORB review are part of any failure, corrective action review on the non-safety-related equipment. The initiation of failure analysis and decisions on corrective action are essentially the responsibility of the maintenance group and any operating or engineering staff directly involved with the particular hardware. The interviews with TMI personnel also indicate it was primarily the maintenance group, though sometimes operations, who would informally recognize they were having a lot of problems with a particular component and initiate a request to engineering to analyze the problem and recommend a solution. A review of work requests indicates failures are mostly fixed by replacing, repairing, or some maintenance activity.

Recent discussions with the TMI superintendent of maintenance indicates TMI is attempting to initiate an equipment failure trend tracking system using the computerized system now available. They are also inputing into the computer the various problems occurring prior to November 1978.

The lack of a system to report, analyze, correct, and bring to management's attention the failures to equipment and procedural errors not considered safety-related was a significant factor in the March 28, 1979, accident.

The staff review of equipment history (reference 56) and the staff report on the condensate polisher (reference 79) clearly show that data was available to management that indicated this equipment was not reliable and had the potential for shutting down the plant and exercising the emergency systems with great frequency. Section III B-5 describes a number of equipment problems that if properly reviewed and analyzed would have led to a safer, more reliable plant. Also a review of the computerized history, even for the short time available, shows that for the equipment that failed or did not work correctly during the TMI-2 event, 87 work requests had been written, only 13 of which were safety-related, which would have potentially brought to bear the overview of TMI and NRC independent review groups.

5. <u>NRC Review-Responsibilities and Performance</u>

In accordance with 10 CFR 21, LERs are submitted to both the region and Washington, D.C., offices of I&E. The region office is the prime recipient for the LER and is the initial contact for any event requiring a 24-hour telephone report.

The regional director is responsible for screening, review, followup, and closeout of all LERs for Region I. He is also responsible for bringing the occurrence to the attention of I&E in Bethesda, Md., if he feels the occurrence warrants such action. I&E may then request technical assistance from DPM, DOR, or DSS, as appropriate. I&E is responsible for review of LERs for generic applicability and design considerations, and for evaluation of the regional program for screening and evaluating events. I&E is also responsible for determining if an event is a potential "abnormal occurrence" that requires reporting to Congress or if the event requires further review or action at headquarters. An important point about the whole flow of LERs through NRC is that LERs are not reviewed in detail unless the region inspector or his supervisor find it necessary to do so (reference 53). Although the region office has primary responsibility for LER closeout, copies of LERs are distributed to the Division of Operating Reactors, the Division of Project Management, the Division of System Safety, and others at headquarters (references 27-30, 62). This is accomplished through the Washington, D.C., I&E office and NRC's Office of Management and Program Analysis (MPA). Under NUREG-0161 MPA is primarily concerned with LERs relative to entry/retrieval of LER data in the NRC computer data base. This computer system provides printouts of LERs -- the LER printout used in this investigation lists all events involving PORV values for the industry from 1968 to date. Although some work has been done at MPA on a more detailed review of LERs -- failure trend data analysis -- such capability is not actually in place and in use. MPA, therefore, is not involved in any in-depth failure analysis/ corrective action on LERs, but rather serves as a data collection, storage, retrieval, and distribution point (reference 96). In fulfilling its role of LER review, I&E can accept the failure analysis/ corrective action provided in the LER by the utility (either the initial or revised LER), or can accept failure analysis/corrective action provided in the LER by the utility after receiving technical assistance from headquarters on failure analysis/corrective action. I&E has this primary responsibility in spite of the fact that it did not participate in the initial NRC design review of the utility nor is involved in design change review.

Thus, I&E lacks a solid background on the design of TMI-2 to support their responsibilities on LER failure analysis/corrective action. The I&E can also turn the LER over to NRR for resolution, but in reference 57, transmitting material requested during Mr. Haverkamp's deposition, Mr. Grier indicated no turnover had been made on TMI-2 LERs from January 1977 to March 1979.

Review of LERs by the Region I inspector is covered in the I&E Manual under procedures 90712B and 90700B (reference 44). No mention is made in the procedure of failure analysis as such, but cause and corrective action are described. Cause (which is the subject of in-depth failure analysis) is to be handled by the inspector as an administrative reporting requirement -- is cause reported accurately and is cause code entered on LER form? Corrective action is handled by the inspector as a technical item -- does technical assessment of corrective action indicate that corrective action is adequate to prevent recurrence? The other technical item assessed under these procedures is safety of operations -- is plant being operated safely under applicable NRC regulations? For significant, safety-related items, the inspector is required to make on-site verification and followup of the LER details. In actual practice, Mr. Haverkamp indicated (reference 17) on-site followup of LERs is limited to those LERs requiring 24-hour notice. The balance (30-day reports) are done on a sample basis (about 5 percent or 10 per year). The items to review and to what detail are determined primarily by the inspector's judgment (reference 17). Likewise, any review of NRC LER data files for similar failures is up to the reviewer and no formal response to the utility is required (reference 53). The judgment of the quality of failure analysis by the I&E regional office depends largely on the reviewer's capability/ background and his perception of himself in these areas -- does he really recognize that the details of an LER are not in the scope of his experience? (reference 17).

As stated previously, various divisions within NRR receive the LER and, therefore, have an opportunity to review it. Discussions were held with each division and depositions from personnel supervising these divisions were reviewed. In the Division of Operating Reactors, all project managers are responsible for looking at all I&E reports on reactors assigned to them.

Each project manager gets a copy of the assent LER. His branch chief sees all LERS for his type reactor. Discussions with these persons indicate, however, they generally do not get involved with detail review of the failure unless requested by I&E (reference 27).

The Division of Systems Safety, which performs most of the engineering/technical review of the construction permit application and operating licensee application, also gets involved in the review of LERs only if asked. Also, they get only the MPA computer printout. DSS has most of the technical or systems specialists in the Office of Nuclear Reactor Regulation, but does not systematically review LERs for applicability to their efforts. (Note: This is a significant point in failure analysis/ integration.) The review of the Davis-Besse LER (discussed previously in section III-B of this report) demonstrates their involvement, or lack thereof, and no further discussion is required here.

The staff review found that the region office and the various NRR offices assumed I&E was evaluating LERs for generic issues. Within I&E, the Division of Reactor Operations Inspection (DROI) has the responsibility for reviewing LERs. In DROI, LERs are reviewed by two people, one for pressurized water reactors (PWR) and one for borated water reactors (BWR). Discussion with these two individuals (reference 98) indicate they do scan the LERs, but rely on the region for detail review and trend analysis. They also thought the licensing organizations were reviewing LERs for generic problems. They also indicated they relied on the region to flag events as there were so many LERs (over 3,000 per year) and significant events were reported through other channels. The DROI director indicated he looks at LERs very infrequently and has known for some time that the review of LERs in DROI is not thorough or complete. He attributes this to a staffing problem. DROI expects the regional offices to detect and correct any missing information in their review of the LER. There is no evidence that DROI makes any in-depth failure analysis/corrective action in its review of LERs (reference 45).

In summary, the staff could not find significant evidence of thorough review of reported events by the various groups that make up NRC. Many groups see the LER, either as written or in summary computerized form. The regional inspector is closest to the problem, but does not do a rigorous failure analysis/corrective action review on all reported events. On nonreported failures, our review indicates all NRC groups are essentially ignorant of the extent of such failures and what the utility is doing to correct them. As discussed in many staff and NRC reports concerning the accident, the lack of a rigorous review of utility failures was one cause of the accident at TMI.

6. Use of LER Data by NRC and TMI

During its investigation, the staff attempted to determine how the LER data available was used by the various NRC and TMI groups to affect what they did and how they did it. As discussed previously in this section, and in section III-B, many people in NRC scanned the reports for applicability, but no significant accumulation of data or evaluation of trends were being made within NRC, and except for I&E, the LER itself was not relied on to pinpoint problems which should be pursued further.

This lack of use was also repeated by the TMI personnel contacted during the interviews on July 6 and 7, 1979 (reference 67). Although copies of the computerized version of the LERs are available to Met Ed people at TMI and Reading, Pa., the large number and lack of specific failure cause and corrective action make them difficult to use. No one person or group of people at TMI had the responsibility to look at LERs in some detail to see if there was a problem they should be concerned with.

When problems did occur, the NRC computer banks could be searched for other incidents of similar nature, but all groups indicated this was not done on a regular basis and might not have been too fruitful due to the poor content of the LER and lack of complete coding for the failure causes in the computer.

Arnold in his letter to Chairman Kemeny (reference 99) summarizes the various sources of event data available to the utility. A large number of sources look at or report events at nuclear power plants in various ways. However, as he points out in the case of the Davis-Besse incident in September 1977, only one, the Nuclear Power Experience, Inc., document, caught the operator's early throttling of HPI and did not identify it as an alert. The primary problem appears to be the quality of the LER that starts the process. As in any computerized system, the old adage of "garbage in, garbage out" applies.

Our review indicates that not only aren't failures being adequately reported, reviewed, and analyzed, but the current output of the NRC LER system is not a significantly useful tool for the utility industry and no other tool is currently available to fill that need. The staff report on the PORV discusses some industry systems that attempt to accumulate failure data, but these too have shortcomings, either because they too use the LER or do not receive data from a broad representation of the industry.

Findings:

- There is no systematic problem reporting rigorous failure analysis, corrective action, problem trend evaluation, and information distribution system applicable to all plant hardware systems, procedures, and operations that are important to plant safety and reliability.

- NRC requirements contained in 10 CFR 21 limit reporting of events by the licensee to essentially those functions and hardware considered safety-related.

- The format and content of licensee event reports as required by NRC do not provide appropriate identification and classification of the problems and their causes; or provide sufficient information for effective utilization by other utilities.

- No NRC organization has had the assigned responsibility to systematically assure a *thorough* review of each LER, the failure analysis contained thereon, the corrective action taken by the utility, <u>and</u> the possible application of the information to other plants.

- There is little evidence of use by NRC or the industry of operating experience or failure history contained in LERs to upgrade requirements, designs, procedures, and training.

E. <u>CONFIGURATION CONTROL</u>

1. <u>Requirements and Procedures</u>

Configuration control may be defined as the discipline that deals with the issuance of an approved plant design and provides a systematic means of controlling changes to that design so that all changes implemented have first been thoughtfully and rigorously analyzed to assure the change is needed, is properly done and verified, associated changes to procedures and training are considered, and changes are communicated to all cooperating groups or individuals. This section reviews the configuration control program required by NRC and implemented by Met Ed for TMI-2 plant modifications and changes that occurred during plant startup and operation periods up to the time of the accident.

This evaluation included a review of LERs' I&E inspection reports, interview transcripts, depositions, and technical staff member reports. The evaluation was further supported by on-site visits to TMI-2, the Met Ed Reading, Pa., office, and the NRC Region I and headquarters offices.

10 CFR 50.59 establishes requirements for change control, tests, and experiments. The regulation permits licensees to make changes within the scope of the Safety Analysis Report and changes to systems not covered by the SAR unless the change involves a change in the technical specifications or an unreviewed safety question.

10 CFR 50.59 states that a proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) if the margin of safety as defined in the basis for any technical specification is reduced.

10 CFR 50, Appendix B, Design Control, requires measures to be established to verify adequacy of design. The requirement establishes that design changes, including field changes, must be subjected to design control measures commensurate with the original design. The TMI-2 design control and related configuration control functions such as design interfaces, field changes and as-built drawings are listed in Section 17.1.3 of the FSAR (reference 61). The quality assurance program for design control is described in Section 17.2.3 of the FSAR. Requirements for the plant and corporate safety committees to review proposed design changes and modifications are defined in Section 6.5 of the technical specifications. Section 5.2.7.2 of ANSI N18.7-1976 specifies that modifications of safety-related structures, systems and components shall be accomplished in accordance with ANSI N45.2.11-1974.

TMI station administrative procedure 1021, Plant Modifications, (reference 100) and generation procedure 1003, control of Design Changes/ Modifications, (reference 101) describe internal TMI requirements and means for approving modifications in accordance with the technical specification and the FSAR requirements. During facility activation and startup, "problem reports" and "field questionnaires" were written to initiate action. These were sent to Burns and Roe and Babcock & Wilcox, if appropriate, for answers or resolution. Where design changes were required, a Burns and Roe engineering change modification was issued to accomplish the work and revise original drawings.

Engineering change modifications (ECMs) are still used if Burns and Roe is involved, but the primary Met Ed system utilizes a "work request" form to record a problem and start work and a "change modification" form to actually delineate and control the design change.

On-site evaluations and discussions indicate that this change "process" at TMI was and can be initiated by most anyone noting a problem (reference 67). The problem may be verbally reported or formally documented such as on a work request or NCR (nonconformance report). The problem is then assigned to an on-site engineer for investigation. Depending on the technical issues and whether the resolution requires a minor or major modification the problem is subsequently assigned to an individual at TMI, to Reading, Pa., engineering, or referred to the architect engineer (Burns and Roe), or to B&W. Assuming a simple problem such as water hammer in a valve with no apparent near-term safety questions, the problem is worked in-house at TMI. In this case, if the assignee determines a surge chamber would alleviate the condition, he initiates preliminary drawings, determines welding processes, etc. The proposed change is then documented on a change modification form and processed to the PORC for review approval, subsequent supervisory, and Reading, Pa. headquarters approval including quality assurance where safety-related issues are involved. Although GPUSC is back on-site for TMI-2 clean up and TMI-1 restart, the basic Met Ed work request change modification system appears to be being used.

2. <u>NRC Review</u>

10 CFR 50.59 requires the licensee to maintain records of changes "to the extent that such changes constitute changes in the facility as described in the safety analysis report. The records shall include a written safety evaluation which provide the basis for the determination that the change, test, or experiment does not involve an unreviewed safety question." Although all changes to safety-related systems are described briefly in the annual report to the region, these requirements essentially result in neither the NRC project manager nor the principal inspector from the region being cognizant of, or participating in, decisions regarding changes to safety-related systems. The utility essentially decides what the NRC looks at. As discussed below the NRC inspector does audit the system to ascertain if the utility makes a proper evaluation of the unreviewed safety question, but a rigorous review is not required or assured. As discussed in section III-B, the project managers in DPM and DOR have the primary responsibility for review of design changes that the utility feels change the license, technical specifications, or involve an unreviewed safety question. The change comes in the form of a request for a change in the license, and the project manager must decide whether detail review is required or not. If the project is still under DPM, technical support would probably come from the Division of System Safety. If the project has been transferred to DOR, then either DOR or DSS technical support might be used. There is no NRC review procedure that requires the original design reviewer to participate in the design change process. DOR representatives confirmed that project management decides what type of review is required and indicated that, at least from TMI-1, information on design changes tended to be light and the TMI review considering need

for NRC approval had been "borderline" (reference 28). In addition, both DOR and DPM personnel contacted indicated they did not overview changes to non-safety-related systems so that in addition to not seeing failures in these systems as discussed previously, they have no change or document control overview that can provide overall utility assessment data.

The requirements and procedures for I&E overview of the utility changes are delineated in I&E Manual chapters 37700B, 37701B and 37702B (reference 44). In order, these basically require the inspector to:

- review one change modification package on each of six systems annually;

- review the completed work on any facility change approved by NRC -- either a license change or an unreviewed safety question; and

- review the change control system once every three years as part of the quality assurance review.

Item one can even be reduced in frequency if no problems are found. These inspection reports procedures are dated Jan. 1, 1979, so procedures prior to that time may have been somewhat different. A number of the inspection reports were reviewed for the one year period prior to the accident (reference 57). These show the principal inspector audited administrative controls for design changes in I&E inspection report 50-320/78-30 on Oct. 4-6, 1978 (reference 26). The report states he spent Oct. 5 and 6, 1978 looking at various operations. That left Oct. 4 for him to review 42 change modification and engineering change modification packages. This review was to verify they were approved and evaluated as required by 10 CFR 50.59. No noncompliances were noted. Obviously, no significant review could be conducted in the time allowed. No indication was given that one or more packages were reviewed in detail.

Apparently, there had been other configuration related inspections conducted in 1977, as two of the reports talk to closing items from 1977 inspections where deficiencies in as-built drawings not maintained up to date (noncompliance 289/77-55-02) and piping supports in variance with detail drawings (noncompliance 320/77-47-09) (reference 57).

I&E is not normally part of the design review process for proposed changes that NRC approves. The region does get a copy of the proposed change, but the project manager has primary responsibility. Haverkamp in his deposition (reference 17) indicated he was not aware of TMI not having an assigned group to update Babcock & Wilcox or other equipment drawings as discussed later in this paper and also indicated he would not be concerned with the configuration of non-safety-related systems like the condensate polisher. In summary, it appears the I&E audits, as constituted at the time of the accident, did not adequately track the changes in progress at the facility, either in quantity or quality, to provide assurance of adequate utility evaluation of changes.

3. TMI

The staff review of the TMI change control system as implemented found both strengths and weaknesses with the latter directly involved in the accident. The basic TMI system to make a change to safety-related hardware is rigorous, appears to meet NRC requirements, and utilizies both on- and off-site review groups including quality assurance and the GRC.

Review of a number of PORC, GRC, and GORB minutes indicates such facility changes are being reviewed though as discussed in section III-B, it is difficult to determine from the minutes the depth of GRC or GORB review. However, the basic system does require both quality engineering and site supervision approval of safety-related changes, and major changes are approved by both manager-generating engineering and manager-generation or quality assurance. Examples of design change documentation were reviewed.

Change modification 849 is a typical package for a change to a safety-related system at TMI. The change was made to feedwater valves FW-V5-A/B and FW-V92-A/B on TMI-1. The changes are described and accomplished in CM 849, and work requests 18,471 and 18,507 (reference 105). The generation memorandum reviewing and approving these changes are included and were reviewed. Quality control surveillance report 79-64 documents the quality assurance surveillance of this change. All documents appear to be in order. The change is approved by the appropriate Met Ed personnel including manager-generation quality assurance. The PORC reviewed the necessary procedure and work requests are approved by all necessary groups including quality control. However, it is noted that this modification was initiated and approved in the January-February 1977 time period. The change is safety-related and was required, based on an engineering evaluation dated Oct. 11, 1979, to provide backup feedwater system isolation capability at TMI-1 within 30 seconds from receipt of signal. The change <u>did</u> <u>not</u> require NRC approval. In spite of the change meeting a safety need, the change was not completed until March 1979. In addition, the quality assurance surveillance report 79-64 indicates there were a number of problems required to be resolved before the system worked correctly and that "Quality Control had elected to spot check the progress of this work using the procedure as a checklist and to witness the post maintenance testing. However, the observance of the post maintenance testing was review because of higher priority surveillance in progress at the time." Although the data showed the system worked properly, the many problems of miswiring, etc., and lack of quality control personnel to provide required surveillance on a safety-related modification is another indication of a less than satisfactory quality assurance program.

A number of changes to non-safety-related systems were also reviewed. Although these do not get independent review, the necessary basic change modification documents are available and signed off by the necessary people.

The licensee is not required to report to the NRC modifications to the plant, even when a safety-related item is involved, so long as the change "does not significantly change the function of that safety-related component" (reference 18). It is the licensee who determines whether it requires a license change or involves an unreviewed safety question. The responsibility of determining reportability of changes has largely been assigned to the TMI PORC with overview by the GRC and the GORB.

A review of PORC meeting minutes indicates discussions are held as to whether or not the change affects the technical specifications or involves an unreviewed safety question. However, of a large number of meeting minutes reviewed on this subject, none provided any technical rationale that could be used to support the PORC decisions. The GORE has had some concerns with PORC review to determine whether an item was reportable. In a memo to Herbein, vice presndent-generation from J. Thorpe, chairman of the GORE on April 6, 1978 (reference 76), Thorpe expresses concern that these decisions are being made without adequate supporting documentation as to why PORC made the determination an item was within license requirements or not an unreviewed safety question. In a note at the bottom of the memo, someone with the initials "JSB", perhaps Bartman, member of GORE, wrote, "Jack, I'm in favor of cutting oon all the 'paper work' we can. If the NRC is not complaining about inadequate documents, I think the GORB should agree it's O.K." This is an example of TMI's feeling toward the NRC. Without technical rationals in the PORC meeting minutes, one can presume that later review of these change decisions by GRC, GORE, or NRC to determine the validity of the decisions would be difficult, if not impossible, to accomplish. This also raises the possibility that the rationale is not always available or considered by the reviewing members.

Although the system to define and control safety-related change appears adequate, the TMI system to assure control of non-safety systems and maintain as-built configuration knowledge of all systems is of major concern. The staff report on the condenstae polisher (reference 79), the system that is believed to have initiated the accident at TMI-2, documents a lack of knowledge by TMI as to how the unit was built and how it is supposed to work. Significant differences exist between the available drawings and the actual configuration of the equipment, and the utility personnel could not explain either configuration. On critical, but not safety-related components like the PORV, decisions are made to add inadequate position indication apparently on basis of cost and without adequate attention of an independent review group or management. A simple analysis would have identified the inadequacies of the added position indicator modification. A staff consultant in his attempt to review the transport of radioactivity from the TMI core to the environs found that up-to-date readable drawings and specifications were not available on-site (reference 133).

A sample of LERs provides additional examples of components being used in other than their intended configuration. LER 78-55/3L, dated Oct. 5, 1978 (reference 106), discusses the control room emergency air handling system damper 4092C failing to close during surveillance testing due to improper orientation of certain instrument air tubing components.

LER 78-51/1T, dated Sept. 6, 1978, and update LER 78-51/1T, dated Sept. 19, 1978 (reference 107), indicates certain safety-related valves did not have cable splices installed per the FSAR, due to being overlooked by the AE at the time splicing modifications were being performed.

LER 78-52/1T, dated Sept. 11, 1978 (reference 108), indicates the main steam lines were determined to be incapable of withstanding a turbine trip from 100 percent power, due to undersized installed restraints which could not suppress faster closure (50 msec) of turbine stop valves than originally specified (150 msec).

LER 78-54/3L, dated Sept. 27, 1978 (reference 109), states the reactor building (RB) sump pump discharge valve to miscellaneous waste holdup tank (WDL-V271) was not supplied with containment grade limit switches, torque switch, and motor, due to a purchasing error by the AE.

A post-accident inspection into the emergency feedwater valve circuitry report in staff analysis report on the closed emergency feedwater valve (reference 80) disclosed that the limit switch was not to drawings and wiring was not to specifications, despite requirement for quality assurance program coverage due to safety-related classification.

A significant part of the problem of configuration control at TMI is the current state of the as-built drawing files. Staff discussions with Met Ed personnel (reference 67) and the recent I&E audit of TMI (reference 55) both document that TMI does not currently meet NRC requirements as delineated in 10 CFR 50 Appendix B, criterion VI and in Section 17.2.11 of the TMI-2 FSAR. Burns and Roe is assigned responsibility for maintaining their drawings up to date, but the large number of engineering change modifications during the latter stages of construction and startup have resulted in a large backlog with updated drawings not being available for months (reference 67). In these same reference interviews, two TMI personnel indicated that no group was currently assigned responsibility to update equipment drawings (B&W, Westinghouse, etc). The master transparent drawings were supposed to be at the TMI site but the manager generation engineering and people at the site were unsure of their status. Currently to determine what a given system is suppose to look like, site personnel must go to three places: the primary aperture card index, a file of outstanding ECMs not yet on drawings, and a similar file of TMI completed change Modification packages. However, as discussed in the staff report on the condensate polisher, this was not enough to produce drawings that reflected the current configuration of the condensate polisher. Further, Richard Vollmer, director of NRC's TMI Support Task Group, testified at pages 25-28 of his deposition that Met Ed's failure to maintain drawings had a time impact on the efforts to devise means for dealing with the accident.

The TMI quality assurance plan in sections V, VI, and XVIII assigns responsibility to:

- each manager for developing and implementing his group's document control procedures;

- the manager-generation operations for implementing the document control system;
- the manager-generation administration for maintenance of the drawing lists;
- the manager-generation engineering for maintenance of the specification lists and updating of drawings;
- unit superintendents to ensure no unauthorized changes are made;
- managers or unit superintendents to ensure document control provisions for vendors; and
- Manager-generation quality assistance for auditing the change control system (once every 2 years).

This diverse assignment of responsibility for configuration control with minimum quality assurance involvement has resulted in an inadequate understanding of the configuration of hardware at the site which may have been directly a factor in the cause of the accident.

Findings

- The limited NRC overview of Utility changes to plant configuration does not assure NRC a current understanding of plant systems and operations.
- NRC personnel involved in the original plant design review during the licensing process are not required to review plant changes.
- I&E personnel responsible for plant overview and acceptance of LER corrective actions are not directly involved in the configuration change process.
- The TMI system for reviewing and controlling changes to "safety-related" systems appears adequate. There is a lack of rigorous control and independent review of hardware configuration and changes thereto for other systems important to plant safety.
- At the time of the accident, TMI did not have a rigorous drawing control system in place that assured plant operators had an adequate understanding of the as-built configuration of all of the facility.
- The I&E inspection program calls for examination of the utility configuration control system once every three years. It did not assure an adequate document (drawings and procedures) control program at TMI.

F. COMPARISON OF THE NUCLEAR ASSURANCE PROGRAM TO THAT OF OTHER SAFETY CRITICAL PROGRAMS

1. Discussion

The disciplines of safety, reliability, and quality assurance have developed primarily over the past 30 years. The need for new management, engineering, and control techniques became apparent particularly in the burgeoning aerospace industry as the safety and reliability requirements of the commercial airplane industry and Department of Defense (DOD) and NASA space programs became more and more stringent as hardware became increasingly complex, the use of environment became more severe, and the number of organizations involved in a given project grew. The Atomic Energy Commission and the Navy experienced similar needs as the nuclear submarine program developed. Commercial industry has also increased their use of and developed these practices as experience indicated such use was cost effective.

In NASA, the development and full use of these techniques is best exemplified in the Apollo, Shuttle, and Centaur Launch Vehicle programs. Apollo and Shuttle were very complex, multiorganization programs. In addition, they were "man-rated" and the capability to return the astronauts safely was of prime importance. As a result, in addition to quality assurance, the engineering technique of redundancy was used, and reliability and systems safety analysis techniques of failure mode and effects analysis (FMEA), hazards, fault tree, and sneak circuit analyses were developed and utilized to identify weaknesses so that corrective action could be applied before operations began.

The Centaur vehicle is one of NASA's primary launch vehicle programs for unmanned spacecraft; weight and payload were particularly critical, and redundancy was generally not possible. In all programs, designing and building the hardware right, understanding the cause of each and every failure, and knowing where each part came from, and on which system it was located, was particularly important as flight failure analysis was generally conducted without the benefit of hardware. The following describes, in general, some of the program's systems, procedures, and techniques utilized by NASA, DOD, and the industry and provides insight into their applicability to the nuclear power plant assurance program. Details regarding these systems, procedures, and techniques are available in the documents referenced herein.

2. Quality Assurance

Current requirements for general quality assurance programs, as used by the NASA and DOD are typified by NASA document NHB-5300.4(1D1) (reference 10) and military specification MIL-Q-9858A (reference 113). A detail comparison of NRC quality program requirements and practices to NASA and DOD program is contained in reference 103. As discussed in this reference, the basic requirements of 10 CFR 50, Appendix B, are very similar to NASA and DOD requirements. Primary differences exist in how these requirements are applied and enforced.

NASA and DOD apply quality assurance to all parts of the program, although depending on the criticality of the component or procedure, some requirements, such as depth of material traceability or the extent of mandatory government inspection may change. NRC, as stated previously in this paper, essentially restricts its quality assurance program to safety-related systems and components and ignores operations and surveillance procedures. As noted previously, the TMI utility followed the same general restriction. In areas of procedures, NASA, for instance, uses quality assurance as an independent review to assure assumptions or requirements the designer specified for use of his hardware are actually included in the operating or test procedure.

A significant difference exists between NRC and utility quality assurance enforcement practices and those of other safety critical agencies. Two primary differences are in use of performance data or trend data on both hardware and organizational performance and the amount of hands-on inspection or participation conducted by both utility quality assurance and NRC enforcement personnel.

In NASA and DOD programs, contractors conduct significantly more quality inspection of its operations as compared to TMI where the quality control staff was hard pressed to do the relatively few inspections required. Contractors are also required to accumulate and analyze nonconformance trend data (reference 10), and utilize this data in their corrective action program. Little use of trend data by NRC or TMI was found in the staff review. The I&E "inspection modules" contain no provisions for measuring performance against quantitative and qualitative standards (except technical specifications) or utilize trending to evaluate utility performance. Both NRC and the utility industry have recognized the problem of accumulating and analyzing failure data for a number of years, but no rigorous system has been developed.

The use of Defense Contract Administrative Services personnel in the plant to conduct mandatory government hands-on inspection and to overview the contractors day-to-day quality, manufacturing, and operations activity is standard practice with NASA and DOD. Until recently, the NRC had no resident inspectors, and except for startup testing, did little hands-on effort.

One other significant difference exists between NASA/DOD and NRC. Section ID501-2 of reference 110 describes quality assurances's responsibility in the design review process and makes quality assurance a vital part of that process. Requirements put on the contractor are duplicated in the agencies where government safety, reliability, and quality assurance personnel participate in the government design review process. Although utility quality assurance is involved in design changes, the inspection side of NRC does not participate in either the original design review or in later changes.

Suppliers of major equipment for both fossil and nuclear power plants recognize the need for a strong quality program to assure the delivery of a reliable product. One example is the General Electric (GE) quality assurance program for large steam turbines described briefly in reference 104. A review of this document shows it not only puts quality into the entire process, but also iterates "product quality can be attained when everyone in the business does his part -- quality assurance cannot be an inspection process only -- quality must be designed and built in, not inspected in."

3. Safety and Reliability

As discussed in previous sections of this paper, NRC does not have a formal requirement for a reliability program and the safety program is based on the design base accidents and the "safety-related" philosophy. Failure mode and effects analysis is not rigorously applied; hazard, fault tree, and sneak circuit analyses are generally not utilized; and the nonconformance reporting system is applicable to only certain systems and activities.

Conversely, the NASA/DOD system, as typified by NASA Handbook (NHB) 5300.4(1D1), section 1D200 and 1D300 (reference 110), have detail requirements applicable to the entire program. Specific requirements include safety and reliability plans similar to the quality assurance plan of 10 CFR 50, FMEA hazards and other safety analysis, a strong closed loop problem reporting, and corrective action system. Of primary importance is the requirement to identify critical potential failure modes or hazards that cannot be designed out and bring these, along with significant problems and failures, to senior management attention for their consideration and resolution.

4. Problem Reporting and Corrective Action

Sections 1D301-6 and 1D506 of NHB5300.41D1 describe the problem reporting and corrective action (PRACA) and nonconformance reporting requirements utilized by NASA for shuttle. Two examples of how such requirements are implemented are contained in references 118 and 119. These systems provide a computerized tracking of all nonconformances and problems and require and allow participation of engineering, safety, reliability, and quality assurance in both the contractor's plant and the government agency in the problem analysis closeout function. The criticality of the nonconformance determines at what level of management a given problem is closed out, but in all cases, engineering plus part or all of safety, reliability, and quality assurance are involved. This differs from nuclear utility and NRC practice where quality assurance is normally not involved in discrepant non-safety-related hardware in the utility and where I&E, which had no part in the engineering design process, is the primary reviewer of problems, failure analysis, and closeout.

It is interesting to note that both the GD/C system run by the NASA Lewis Research Center and the PRACA system run by the Johnson Space Center have computer programs in place that are capable of remote input and output of specific or generic failure and status data and allow ready access by all organizations involved in the process. Similar systems can be used by NRC. The GIDEP/Alert System (reference 114) is another system in use in the aerospace industry for dissemination of generic problems. Its techniques are also applicable to the nuclear power industry.

5. <u>Safety and Reliability Analysis Techniques</u>

Previous sections of this paper have discussed the various reliability and safety analysis techniques. The techniques of failure mode and effects analysis, hazards analysis, and sneak circuit analysis are described in reference 121. Other methods are available and used, but these are probably the most appropriate to the nuclear power plant as it exists today. Much discussion has taken place in other sources about the use of probabilistic modeling in reliability analysis. This may have some usefulness for certain applications in the nuclear industry, but the current nuclear program is much like Apollo and Centaur were in that there is insufficient data on an insufficient number of units to do an effective mathematical modeling analysis.

In addition to the consideration as to whether a FMEA or hazard analysis is done, or not, the depth of the FMEA is also significant. Section 1D301-3 of NHB5300.4(lDl) describes the detail required and indicates the FMEA is conducted to the black box or what is sometimes called the "line replaceable unit" level. Where failures of these components are critical, further analysis is conducted to the piece part level. While such depth may not be required in nuclear power plants, until a basic analysis is conducted, the criticality of a component cannot be assessed.

6. <u>Management Involvement and Attention to Detail</u>

The staff review of the NRC and utility independent assessment program has identified a number of weaknesses associated with government and utility management involvement in, or use of, the independent assessment program and <u>the lack of attention to detail</u> in all systems and functions. Two documents available to the Commission, references 111 and 112, document the importance these activities were and are to the NASA Apollo and Naval Reactors program. Both Low's statement to the Subcommittee on Energy Research and Production of the Committee on Science and Technology (reference 111) and Rickover's comments to the Presidential Commission (reference 112), stress the need for top management's involvement in the overview and problem resolution process. Though each program has its own particular characteristics and the particular management system to assure success may differ, the general philosophy still exists. To quote Rickover, "Reactor safety requires adherence to a total concept wherein all elements are recognized as important and each is constantly reinforced." He also discusses the need for decision makers to have a fundamental understanding of the technical aspects and assure that careful attention is paid to technical detail. Also, to quote Rickover, "Managers must get out of their offices and see what is really going on."

Both Rickover and Low stressed the need to consider the operator in the design process. In the case of the Navy, the objective is to make the unit "sailor proof" and assume the operator can err. In the case of Apollo, Low stressed the need to include the operator in the design review process. This emphasis also applies to the nuclear power program.

Low also stressed the need to do rigorous failure analysis and corrective action review and the need to analyze fully the "what ifs." Rickover stressed operator selection, qualification, and training. This emphasis also applies to the commercial nuclear power program.

Involving management into the program processes is not an easy task and requires a desire of management to be involved and to use the tools of safety, reliability, and quality assurance to support that involvement.

Following a number of Centaur Launch Vehicle failures in the late 1960s, NASA and General Dynamics evolved a "Mission Assurance Program" (reference 120) which integrated management and the various assurance functions into an overall management plan. It illustrates one method of accomplishing the objective of assuring all decisions, design, fabrication operations, failure closeout, etc., are adequately reviewed and decided at the right management level. At the same time, the system gives management the visibility to be able to recognize weakness in the program through audits and trends so that problems (management, hardware, or operations) can be addressed before they have a critical impact on that program.

As described in this section, differences in the procedures and practices of the NRC/utility nuclear power program and aerospace/naval reactor programs, illustrate many of the weaknesses currently ingrained in the nuclear power program. Correction of these weaknesses can provide the basis for a safe nuclear power program. As Low said, "Safety cannot be forced from the outside, it must come from within."

Findings

- The overall quality assurance, safety, and reliability programs and practices utilized by NRC and GPU/Met Ed are not commensurate with the requirements, procedures, and practices of other programs where safety and reliability are critical concerns.

- Management, engineering, quality assurance, safety, and reliability practices and philosophies are available to minimize the probability of failures in the nuclear industry.

IV. FINDINGS AND CONCLUSIONS

A. SPECIFIC FINDINGS

For the convenience of the readers, the specific findings are listed here in the same order that they were identified in the text. The most important parts of the specific findings have been combined into the major findings that are listed in the conclusions section of the report.

The specific findings, by report sectioc., are as follows:

Requirements

- Quality assurance requirements as stated in 10-CFR-50 Appendix B, appear adequate for those systems to which they apply.
- Quality assurance requirements apply only to a narrow portion of the plant defined as safety-related or safety grade. Many items vital to the safe and reliable operation of the plant are not covered by the quality assurance program because of this definition.
- There is no requirement for independent, on-site quality, or safety assessment operations. Surveillance testing by the utility is audited infrequently. Regulations allow review to be done by in-line supervision and other personnel directly responsible for operations.
- Reliability/safety analysis requirements are applied to specific safety-related hardware as specified in Appendix A of 10-CFR-50 utilizing a questionable "single failure" criterion.
- Safety and reliability requirements and analyses are not required to be applied to many plant systems which may be "vital" to the safe operation of the plant, but are not labeled "safety-related."
- Lack of requirements by NRC in the safety and reliability disciplines has resulted in little motivation to form a strong safety and reliability engineering capability in NRC and the utility industry.
- Present NRC design, safety, and reliability requirements do not generally address human factors and the man-machine interface.

NRC Organization and Responsibilities

- There is no assignment within the NRC organization for overview of critical functions such as: problem reporting, failure analysis, and corrective action; systems engineering; and the role of the operator and human factors in plant safety.

- The fragmenting of quality assurance responsibilities among the various NRC organizations weakens the ability of this discipline to ensure an adequate utility quality program.

- The NRR Division (DOR) responsible for overviewing the operating reactor is not part of the licensing design review, construction, or startup monitoring process.

- No NRC organization is identified as being responsible for auditing the project management, engineering, and inspection functions of the NRC.

- NRC project managers and quality assurance personnel in the NRC Division of Project Management and Operating Reactors are primarily concerned with initial licensing and changes thereto within the scope of the FSAR and Standard Review Plan. Little overall assessment of utility management, engineering, or operations is evident.

- The NRC project manager does little engineering analysis and is not a significant factor in the review of nonconformances, procedures, or system engineering aspects of the plant.

- Project management experience gained during design construction and startup of the plant is lost upon transfer of responsibility for the plant to DOR. There appears to be little effort by the project manager in DPM to transfer licensing and startup experience to other NRC groups.

- There is no NRR review of proposed operating procedures as part of operating license approval.

- The Division of System Safety overview of the nuclear power plant is primarily concerned with the design of safety-related components and subsystems within the framework of the Standard Review Plan.

- The DSS does not include nor does the Standard Review Plan require significant consideration of non-safety-related systems, systems interactions, operating procedures, or human factors in the evaluation of the nuclear plant.

- The DSS has not adequately recognized potential system and system-operator problems even when these problems were brought to their attention; possibly because of the emphasis applied to component and subsystem design aspects and to the design base accidents by the NRC.

- The DSS makes little use of plant experience data in developing requirements for and in the conduct of their overview process.

- The NRC Office of Inspection and Enforcement and its regional office conduct a detailed, documented inspection program for those utility systems and activities covered by applicable regulations, regulatory guides, utility FSAR, operating license, and technical specifications.

- Region I on-site inspections appear to miss signals and symptoms that indicate potential plant operating problems and weak utility management.

- In Region I, there is little physical inspection or direct observations of operations such as surveillance testing of the operating reactors during NRC plant visits.

- Region I inspections did not detect the emergency feedwater valve procedure change leading to technical specification violation in about 15 visits to TMI-2 from August 1978 to March 1979.

- The role of quality assurance does not appear to be an important factor in the I&E plan. No I&E audit was made of the TMI-2 quality assurance plan to see that that plan was implemented to support the operating phase from the beginning. An I&E audit about 18 months after operating license issuance found many deficiencies in the implementation of the quality assurance plan. In their investigation of the TMI accident, I&E did not interview any Met Ed quality assurance personnel in the 200 interviews held.

- Sufficient I&E staff may not be available to conduct an adequate overall plant surveillance (inspection) activity.

- There is little I&E assessment of the utility's management capabilities.

- Although one inspector receives all reports concerning TMI-2, he has no responsibility for the execution or the quality of execution of all TMI-2 sections.

<u>Utility (Met Ed) Organization and Responsibilities</u>

- The Met Ed organizational structure, quality assurance plan, and independent review groups meet basic NRC requirements.

- As implemented, the TMI independent assessment program involving quality assurance and the review committees, PROC, GRC, and GORE, looked only at NRC required safety-related functions and therefore could not assure safe operation of the overall plant.

- Lack of quality assurance or other TMI independent assessment of non-safety-related hardware and procedures was a factor in the accident.

- Because of the limited purview of the review mechanisms, it is possible that Met Ed management was not fully cognizant of plant conditions and operations.

- Although the TMI internal audit program meets NRC requirements and is well done, Met Ed management did not assure that corrective action identified by the audits was initiated and completed in a timely fashion.

- Significant misunderstanding exists among NRC and TMI-2 personnel regarding the meaning and application of terms such as "safety-related," and "safety grade," and similar terms.

- Misunderstanding exists among NRC and TMI management and project personnel as to what specific hardware is considered safety-related at TMI-2 and what specific document defines that hardware.

- The lack of clear designation of safety-related equipment and, specifically, what that means contributed to inadequate hardware and procedure review and failure analysis and corrective action that are necessary to assure safe operation of the plant.

Procedures

- There is essentially no NRR review of detailed utility procedures. Reviews are limited to assuring that a proper list of procedures is available and a utility procedure review system is in place.

- I&E review of procedures is limited by intent to about 5 percent of operating and emergency procedures, and changes to procedures identified by the utility as impacting the technical specification.

- The PORC is the primary procedure review organization. Current PORC membership and review practices appear to preclude adequate independent review of procedures associated with safety-related systems.

- Lack of TMI quality assurance overview of the preparation and conduct of surveillance procedures can preclude detection of omissions, mistakes, and unsafe practices by the utility.

- A small utility quality control staff precludes adequate verification (inspection) of maintenance and repair of safety-related systems and components.

- There is no independent review or verification of maintenance and repair procedures involving systems not identified as safety-related, but which may be important to safe and reliable plant operations.

Nonconformance Reporting Systems

- There is no systematic problem reporting, rigorous failure analysis, corrective actio., problem trend evaluation, and information distributing system applicable to all plant hardware systems, procedures, and operations that are important to plant safety and reliability.

- NRC requirements contained in 10-CFR-21 limit reporting of events by the licensee to essentially those functions and hardware considered safety-related.

- The format and content of license event reports as required by NRC do not provide appropriate identification and classification of the problems and their causes; or provide sufficient information for effective utilization by other utilities.

- No NRC organization has had the assigned responsibility to systematically assure a thorough review of each LER, the failure analysis contained therein, the corrective action taken by the utility, and the possible application of the information to other plants.

- There is little evidence of use by NRC or the industry of operating experience and failure history contained in LERs to upgrade requirements, designs, procedures, and training.

Configuration Control

- The limited NRC overview of utility changes to plant configuration does not assure NRC a current understanding of plant systems and operations.

- NRC personnel involved in the original plant design review during the licensing process are not required to review plant changes.

- I&E personnel responsible for plant overview and acceptance of LER corrective actions are not directly involved in the configuration change process.

- The TMI system for reviewing and controlling changes to safety-related systems appears adequate. There is a lack of rigorous control and independent review of hardware configuration and changes thereto for other systems important to plant safety.

- At the time of the accident, TMI did not have a rigorous drawing control system in place that assured plant operators had an adequate understanding of the as-built configuration of all the facility.

- The I&E inspection program calls for examination of the utility configuration control system once every 3 years. It did not assure an adequate document (drawings and procedures) control program at TMI.

Assurance Functions of Nuclear Power as Related to Other Programs

- The overall quality assurance, safety, and reliability programs and practices utilized by NRC and GPU/Met Ed are not commensurate with the requirements, procedures, and practices of other programs where safety and reliability are critical concerns.

- Management, engineering, quality assurance, safety, and reliability practices and philosophies are available to minimize the probability of failures in the nuclear industry.

B. CONCLUSIONS

A review of the independent assessment program for nuclear power plants as defined by NRC quality assurance regulations and requirements has been accomplished by examining the major elements of the NRC and one of its five regional offices and one utility company (Met Ed). This somewhat limited review has resulted in two general conclusions and several major findings. The major findings relate to the specific findings listed in the previous section of this report. The findings are supported by the results of the analysis by the Department of Energy (DOE) (reference 122).

It is concluded that the overview and independent assessment performed by NRC were limited only to those items which were identified as safety-related, including intensive analyses of recovery from postulated accidents which resulted in a narrow overview of the utility. This narrow view was further confined by the application of the forerunner of the Standard Review Plan which programmed the review effort by NRC to carefully defined areas. Further, this narrow and confined review was bothered by a focusing problem brought about by doubts about the interpretation and application of the term "safety-related" to equipment; this further affected related procedures, inspection, maintenance, and problem resolution. Combining this narrow view with a weak NRC-to-utility management interrelationship, left voids that prevented the NRC from knowing the "health" of the utility. More important, the NRC did not have an independent assessment activity to "tell them that they didn't know."

It is further concluded that the management utility joined the NRC's narrow and confined view on the safety items and virtually ignored other vital parts of plant operation. This viewpoint is shared in an analysis by DOE (reference 123). These other parts were those whose performance not only supported the safety-related items, but were those that were also vital to assuring that the plant would reliably perform. This illustrated that the utility management had not exhibited the desire or capacity to go beyond the NRC requirements to provide a well-designed, maintained, and staffed plant capable of reliable performance that would not jeopardize the health and safety of the public and its own workers. Like the NRC, the utility management had no independent assessment system to tell them that their plant was "sick."

The major findings are as follows:

- The NRC organization, procedures, and practices, as now constituted, do not provide for the combined management, engineering, and assurance review of utility performance necessary to minimize the probability of equipment and operator failures necessary to ensure the safe operation of the nuclear power plant.

- A lack of an independent on-site quality assurance or safety assessment of plant operations and of equipment not considered safety-related contributed significantly to the accident at TMI.

- There was lack of detailed safety and failure modes analysis on all plant systems necessary to ensure the reliability and safety of the facility.

- Systems engineering, interactions between systems, and the interaction between the equipment and its operators have not generally been considered in the NRC overview process.

- A comprehensive nonconformance, problem reporting, failure analysis, corrective action system applicable to all systems and operations that affect plant safety and reliability does not exist. The current LER system also does not assure adequate dissemination and utilization of useful failure data through the industry.

- Current utility and NRC practices do not assure proper preparation, review, and execution of operating and maintenance procedures.

- NRC has a very limited view of changes made to plant configuration. Utility control of safety-related equipment changes appear adequate; control of non-safety-related equipment configuration is inadequate.

- Full use is not being made of management, engineering, safety, reliability, and quality assurance practices which are in use in other industries where safety and reliability are critical concerns.

ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
AE	Architect Engineer
AEC	Atomic Energy Commission
ANSI	American National Standards Institute
ASLB	Atomic Safety and Licensing Board
B&W	Babcock and Wilcox
CFR	Code of Federal Regulations
DCAS	Defense Contract Administrative Services
DOE	Department of Energy
DOR	Division of Operating Reactors
DPM	Division of Project Management
EPRI	Electric Power Research Institute
ERDA	Energy Research and Development Agency
FMEA	Failure Mode and Effects Analysis
FSAR	Final Safety Analysis Report
GORB	General Office Review Board
GPUSC	General Public Utilities Service Corporation
GRC	Generation Review Committee
I&E	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronic Engineers
LCVIP	License Contractor and Vendor Inspection Program
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
MC	Inspection and Enforcement Manual, Chapter
Met Ed	Metropolitan Edison Company

MPA	Office of Management and Program Analysis
MGQA	Manager of Generation-Quality Assurance
NASA	National Aeronautics and Space Administration
NHB	NASA Handbook
NMSS	Office of Nuclear Material and Systems Safeguards
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PA	Product Assurance
PORC	Plant Operating Review Committee
PORV	Pilot-Operated Relief Valve
PRACA	Problem Reporting and Corrective Action
QAB	Quality Assurance Branch in Division of Project Management
QA	Quality Assurance Program
QC	Quality Control
R&D	Research & Development
RES	Office of Nuclear Regulatory Research
R&S	Reliability and Safety
RSB	Reactor Systems Branch
SAR	Safety Analysis Report
SD	Office of Standards Development
SER	Safety Evaluation Report
SRP	Standard Review Plan
SR&QA	Safety, Reliability, and Quality Assurance

METHODOLOGY

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REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

CONDENSATE POLISHING SYSTEM

BY

William M. Bland, Jr.

October 1979
Washington, D.C.

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SUMMARY

On March 28, 1979, the Three Mile Island-2 nuclear power plant experienced the most severe accident in the U. S. commercial nuclear power plant operating history. The accident, which occurred at about 4:00 a.m., was started by a loss of normal feedwater supply to the steam generators which led rapidly through a normal shutdown sequence of events for reactor shutdown. During this normal sequence the pilot-operated relief valve (PORV) opened and stuck (failed) in the open position, in which it remained for a considerable length of time before being isolated by action of the operators. This failure, followed by early operator action that throttled the HPI (high pressure injection) pumps, initiated an abnormal sequence of events that led to this accident. The significant findings and a conclusion are listed below:

<u>FINDINGS</u>

1. The condensate polisher effluent valves closed at the beginning of the accident on March 28, 1979.

2. These valves had closed unexpectedly twice before under similar surveillance procedures, and again later under a related condition.

3. Tests at TMI-2, to date, have not confirmed a reason for the closure. A possible reason has been postulated by an analysis which shows that water accumulating in the control air line could trigger valve closure.

4. The condensate polisher as used at TMI-2 had essentially a zero operational margin.

5. The polisher bypass valve is not designed for automatic emergency opening nor even for manual opening in an emergency, although its availability is noted in the Final Safety Analysis Report (FSAR). If appropriately designed, this valve could provide substantial operational margin.

6. The condensate polisher bypass valve failed to open by remote control during the accident, as it had at least once before; corrective action had not been accomplished.

7. Resin removal from polisher has been a chronic problem,and has existed since early system tests. Corrective action has been ineffective. Removal problems appear to contribute to unexpected closing of effluent valves. Design problems were not worked out with the polisher designer.

8. Available drawings at TMI are significantly different from the condensate polisher. No interface drawings or integrated schematics are available for use by procedure writers, maintenance people, or system engineers.

9. Condensate polisher is not classified as "safety-related;" did not receive detailed design analyses; did not receive quality control coverage during operation; did not receive management and management review group attention for problems with hardware or for procedure control.

10. There is a history of operational problems and a large amount of maintenance work on the polisher.

CONCLUSION

The condensate polisher, although vital to the operation of the plant, did not receive appropriate attention in design and from assurance function, engineering, management, and management review groups; proper attention could provide a significant increase in plant reliability.

INTRODUCTION

On March 28, 1979, TMI-2 of the nuclear power plant at Three Mile Island near Middletown, Pa., experienced the most severe accident in the history of commercial nuclear power plant operations in the United States. This accident has been and continues to be the subject of much investigation in order to determine the primary cause behind the initiation and furtherance of the accident.

A number of analyses, involving studies, inspections, and tests have been conducted to understand what caused this accident and why it happened. One of these analyses -- that of the condensate polishing system -- is described in this report.

This report, which is really more investigative than analytic in nature, has been prepared to present the facts that can be gathered that relate to one part of the accident and to examine these facts to see what lessons can be learned that can be used to possibly prevent a recurrence of the March 28, 1979, accident.

The function of the condensate polishing system, which is described in reference 1 with other parts of the TMI-2 nuclear power plant, is to maintain water quality by removing impurities from the condensate -- the objective being the prevention of problems in the power conversion system caused by scale formation, corrosion carryover, and caustic embrittlement. In addition, the system design provides for removing impurities in the condensate caused by inleakage in the steam generator of reactor coolant liquid, and intermittent inleakage in the condenser of cooling water from the circulating water system. The design also provides a bypass of the entire condensate polishing system. The design of the polisher units and regeneration equipment is based on a 28-day period. The system is so designed that seven of the polisher units can handle the full condensate flow while the remaining one is being replenished.

The equipment in the condensate polishing system is in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections VIII and IX. However, it is not classified as "safety grade" and thus does not receive the same care and attention from quality control during its operational life, nor did it receive such care when it was designed, fabricated, transported, stored, installed, and checked out.

The condensate discharged from the condensate pumps enters a header in the condensate polishing system and is routed through seven of the eight polishers; the water flows down through the polisher resin bed and out to service. Each polisher contains a mixed resin bed consisting of <u>anion</u> <u>cation</u> <u>ion</u> exchange resins. There are nine resin charges in the polishing system: one charge in each of the eight polishers, and the ninth charge in either the receiving tank or the mixing and storage tank. Under normal operating conditions, there is a flow of 2,500 gallons per minute (gpm) of condensate through each of the seven in-service polishers; the eighth polisher is on standby. Each polisher remains in-service until its resin bed has been exhausted--when the polisher is no longer providing effluent of the desired quality.

Radioactivity will appear in the condensate polishing system if a leak occurs in the steam generators allowing primary coolant to leak into the secondary system.

ANALYSIS

RELATED ACTIVITY JUST PRIOR TO THE ACCIDENT

As reported by two sources (references 2 and 3) and as noted in various interviews, depositions, and hearings, the TMI-2 plant operating staff had been working on the polisher for some time to clear resin from polisher tank 7 when the accident was initiated. This work was reportedly being accomplished in accordance with Operating Procedure (OP) 2106-2.2, March 21, 1979, and had been in progress for about 11 hours prior to the accident. The work involved the use of compressed air and water, as per the procedure to force the spent resin from the tank.

At the time the turbine trip was announced, an operator reported that the condensate polisher panel indicators showed condensate polisher isolation, which indicated no flow (reference 2) through the polisher. This condition could be caused by closed polisher effluent valves. This state of no flow at this time was confirmed through a review of records by a Nuclear Regulatory Commission (NRC) investigator (reference 2), and by this review, Appendix J.

Finding

The condensate polisher effluent valves closed at the beginning of the accident.

REPORTED POST-ACCIDENT INSPECTION RESULTS

A number of items that can be classified as nonconformances, or things out of the ordinary, have been reported as found on the Polisher after the event. The reported items are as follows:

- Water in the instrument and service air receiver tanks (references 2 and 3).
- Check valve in service air line stuck open (reference 2).
- Failure of remote opening of condensate polisher bypass valve (reference 2).
- Effluent valves control solenoids improperly wired (references 2).

Without discussion, these items convert to findings, as follows:

- Water was found in the air systems.
- Polisher bypass valve failed to open.

PRE-ACCIDENT RELATED EXPERIENCE

Experience at TMI-2 relative to the capacity of the instrument air system and problems with water in the air system is recounted in reference 2. From this discussion, interconnection of the air systu&s, the instrument air systems, and the service air system to provide adequate capacity may not have been a wise decision because of subsequent concerns and modifications. For instance, William Zewe's note of May 15, 1978, Appendix A, recommended modifications and contains a note of concern, as evident by the following quotation:

> It's time to really do something on this problem [subject of note is water in service air and instrument air] before a very serious accident occurs. If the polishers take themselves off line at any high level of power resultant damage could be very significant.

Zewe's note called for making the polisher bypass valve automatic. That this request was not implemented is confirmed by inspection and the valve problems during the TMI-2 accident. Apparently, it met the same fate as an earlier request of Nov. 3, 1977, to improve the operating performance of the bypass valve, which was disposed of by the Met Ed test superintendent on Nov. 17, 1977, as follows:

> No further action required by this PR (Problem Report). If, when the plant is restored the problem is better defined, we will resolve the problem.

An even earlier (Nov. 14, 1977) memorandum, "Water in the Instrument Air Lines at the Condensate Polisher Control Panel and Regeneration Skid," discussed a loss of feedwater condition in TMI-2 on Oct. 19, 1977, (also contained in Appendix A) by Brummer and Ross, and recommended extensive modification to prevent recurrence of that event. The description of the Oct. 19, 1977, event shows close similarity to the events believed associated with the polisher on March 28, 1979. To illustrate the similarity, a descriptive paragraph from the November 1977 memorandum is given below:

> During or shortly after the attempted transfer of resin from the mix bed polisher #3 to the receiving tank on the regeneration skid, the Auxiliary Operator noted water running out the air operated recorders on the condensate polisher control panel, No. 304. Shortly thereafter the discharge valves on the condensate polishers closed resulting in a total LOSS OF FEEDWATER condition. Upon detection, the Control Room Operator immediately tried to open CO-V12, condensate polisher bypass valve; however, he was unable to open this valve from the control room. The auxiliary operator was then notified to manually open CO-V12. After about 60 minutes, and assisted by another Auxiliary Operator, CO-V12 was opened. If this would have happened while at power the unit could have been placed in a severe transient condition resulting in an Emergency Feedwater Actuation, Main Steam Relief to Atmosphere, Turbine Trip and Reactor runback with possible trip.

Still earlier problems in removing the spent resins in the polisher, as expressed in Field Questionnare 1577, February 9, 1977, resulted in modifications to the polishers; modifications were apparently completed about April 20, 1977, Appendix A.

On May 12, 1978, as indicated from a review of TMI-2 chemistry-polisher log that is maintained by technicians to record normal activity with the equipment (Appendix J), the effluent valves of the condensate polisher closed unexpectedly. At the time of closure, the log reports the operators were changing out polishers. Polishers 6, 7, and 8 were involved, with tranfer of resin from 7 being done at time of the event. Water was found in the air lines. This event was not reported in the MOR (monthly operating report) for May; the plant was down for other work at the time. By interpretation of requirements for the MOR, significant operational events are to be included. In this case, the plant was already down, so this event was not considered to be significant.

This briefly treated history of problems with the TMI-2 condensate polisher indicates problems that appear to have been chronic relative to removing spent resin, water in the air supply, and reluctance of the polisher bypass valve to operate. Reports had been made, and at least some corrective action had been performed. However, from what is believed known about the polisher's role in the March 28 accident, each of these three main concerns were present.

Another incident of TMI-2 tripping because of a loss of feedwater is reported in reference 2 and Appendix A-1:

> . . . on November 3, 1978, the unit tripped from 90 percent power on high pressure due to loss of feedwater. This occurred when an instrument technician mistakenly opened the control power supply breaker to the condensate polisher control panel causing all polisher outlet valves to close.

<u>Findings</u>

- The condensate polisher effluent valves had closed unexpectedly in 1977 and in 1978 when resin was being transferred from a polisher tank with water and air, as was being done on March 28, 1979.
- Design deficiency in the bypass valve was not corrected.
- Problems in removing spent resin remained unresolved.
- Problems were not thoroughly investigated and analyzed.
- Continued problems were experienced with water in the instrument and service airlines.
- Recommended corrective actions were not thoroughly considered.
- Corrective actions were not verified as being able to accomplish intended action before use.

- The condensate polisher effluent (outlet) valves closed in 1978; while unit was at 90 percent power, the electrical power to condensate polisher control panel was interrupted.

DESIGN

The TMI-2 condensate polisher was originally designated for use at the Oyster Creek facility (Appendix P), where it was to be required to process 100 percent of the condensate flow. On relocation, the changes made in the design were the minimum required to accommodate the geological differences of the TMI site.

Possibly the most important deficiency is that the TMI-2 polisher did not have designed into it an automatic fast-acting bypass valve that could be utilized in the event of flow problems through the polisher. Also, from the reports of investigation into the bypass valve, Appendix P and Appendix J, and the photographs in Appendix J-3, it can be seen that the valve was not designed for emergency manual actuation. Its location and orientation preclude rapid access and manual actuation. However, the content of the TMI-2 Final Safety Analysis Report (FSAR), page 10.4.8, reference 1, implies that a working bypass exists: "the design also provides a bypass of the entire condensate polisher system." As noted in Appendices M and N, there have been some differences of opinion in the design objectives in the TMI-1 and TMI-2 condensate polisher bypass capabilities. However, use of the bypass valve, to carry some of the flow in parallel either with the condensate polisher or with a working automatic valve could provide an operational margin that would make the plant less sensitive to trips caused by perturbations in the condensate polisher outputs.

This is substantiated by the study conducted on equipment conservation (Appendix Q), which included the condensate polisher as one of three types of equipment analyzed. It was reported that the condensate polishing system as designed and used at TMI-2 had essentially a zero performance margin when the plant was operating at rated power. The analysis further indicated that appropriate use of a properly designed condensate bypass valve could increase the performance margin significantly. The appropriateness of this analysis is strengthened by a limited survey made of other nuclear power plants' use of condensate polisher bypass valves. The results (shown in Appendix 0) indicate that there are a number of different designs, including, for example, the recent use by the Crystal River plant of an automatic bypass valve that saved a plant trip.

It also utilized a deep bed demineralizer instead of the precoat type, as in TMI-1. The type used in TMI-2 has had a number of handling problems, as noted in the previous section in the discussion of the Oct. 19, 1977, and May 12, 1978, loss of feedwater event; in the problems in removing spent resins, Feb. 19, 1977; and in the problems apparently associated with removing resins from polisher 1 on March 22-23, 1979, as discussed in the next section and again on March 27-28, 1979, reference 2. From a review of the polisher log, it has been noted that problems in transferring resin from polishers occurred frequently, about one out of every 12 transfers (Appendix J). This is an indication of a chronic problem.

It is noted that the condensate polisher is not classified as safety-related equipment (appendix P, and Appendix F). This meant that it did not receive considerations in its design of requirements for analyses and tests as in QA Program. As noted in Appendix Q:

> although the condensate polisher is vital to the turbine operation and plant output, no analysis was performed nor requirements formulated to insure that sufficient active polisher legs would be in operation to adequately feed the condensate booter pumps at all operational flow conditions.

It is also noted in Appendix Q that no design studies or analyses were performed to identify the various failure modes of the polisher and the effects of the modes on plant operation.

The supplier of the condensate polisher -- L. A. Water Treatment, -- has produced a number of polisher systems (Appendix L), including nine for nuclear power plants other than TMI-2. Reviews of the design process followed has not disclosed any other significant problem that can be related to the accident until 1971. Earlier problems appeared to have been coordinated with the equipment supplier and resulting changes show up in the supplier engineering drawings. Subsequent to about 1971, the coordination of problems for analysis and corrective action does not appear to have included the services of the equipment supplier (Appendix K and Appendix Q). Six field changes applied since 1971 are listed in Appendix K; two significant ones are changes related to problems of removing resin from polishers.

During the post-accident review at TMI with condensate polisher operators and system engineers from GPUS, it was noted that the available drawings were significantly different from the actual condensate polisher equipment (Appendix J). For example, valves were shown in wrong locations and had improper identification; components were shown in incorrect locations; and air lines locations and interconnects were not properly shown. It also was noted that interface drawings and integrated system schematics were not available. Drawings have been marked up to show examples of changes needed to make the drawing the same as the polisher equipment (Appendix J-2).

Without accurate drawings, interface drawings, and integrated schematics, operators, maintenance personnel and procedures preparers have difficult, if not impossible, tasks to do.

Findings

- The condensate polisher was not classified as "safety-related" equipment, and thus it did not have design analysis and tests required in Quality Assurance (QA) Program applied.
- The condensate polisher as used at TMI-2 had essentially a zero operational margin.
- Use of an appropriately designed bypass valve in parallel with the polisher could have provided a substantial operational margin.

- The condensate polisher bypass valve was not designed to automatically actuate upon demand, nor was it designed for emergency manual actuation; although apparently considered so in the TMI-2 FSAR.

- There were problems associated with resin removal; problems developed about one time in each 12 transfers.

- Design changes were not coordinated through the designer after 1971.

- Available system drawings were significantly different from equipment; neither interface drawings nor integrated system schematics were available.

PROCEDURES

From reference 2 it is believed that the condensate polishing system Operating Procedure 2106-2.2 (Appendix B, Aug. 21, 1979, revision 9) was being used by the plant operating staff just prior to the accident on March 28, 1979. That this revision of the procedure was being used is supported by comments in Appendix C; however, later communications (Appendix D) indicate some uncertainty as to whether revision 9 or revision 8 of the procedures was being used.

The procedure is very long and provides for the positioning of many valves in precise order and alignment although the procedure is arranged so that the steps can be checked off when completed. Statements in reference 2 indicate that progress had been made most of the way through the procedure at the time the accident was initiated. This is confirmed by Appendix D, which indicates that progress through the procedure stopped between steps 4.1.4E5 and 4.1.5E6, for either revision of the procedures. This step involved the use of water to transfer the resin.

Revision 9 of this procedure had been extensively revised on March 20, 1979. The earlier procedure, revision 8, is included as Appendix E. The procedure change request is included in Appendix F. The reasons given for the change were, "to incorporate new acid procedure, new short regen procedure, new trouble shooting section and additional operator guidance and instruction." The changes appear to affect the majority of the pages in the procedure, although some only slightly, others more so.

The change action was initiated on Jan. 25, 1979, and the revised procedure was approved on March 21, 1979, when it was signed by the engineering supervisor and the TMI-2 superintendent. It did not require the approval of the Plant Operations Review Committee (PORC) because it was not classified as a nuclear "safety-related" procedure, or require quality assurance, and did not show verification before use. It cannot be established which revision of the procedure was in use on March 28, thus indicating a lack of document control and work control.

Excerpts taken from the condensate polisher log for TMI-2 (Appendix G) indicate that polisher 7 was worked on in the period March 22-23, but some problem was apparently encountered, and it was put back in service before being completed. The problem that was encountered is not known. Another indication that there was a problem on March 22-23 is that the polisher was again started through the procedure on March 27 (the normal time between work on a given polisher is about 28 days, reference 1).

Findings

- Control of surveillance procedure was lax.
- A recently revised surveillance procedure may have been used in the TMI-2 condensate polisher on March 28.
- It is not clear that the revised procedure was verified before being put into use.
- There were difficulties encountered with polisher 7 before March 28 and on March 28.

RECORDS SEARCH AND QUALITY CONTROL

All documents used to record equipment problems at TMI-2 have been searched. The files of work requests, equipment history cards, and nonconformance reports at TMI were searched; the NRC licensee event report (LER) files were searched; and sampling was conducted on the files of problem reports, filed questionnaires, discrepancy report, and unsatisfactory and inspection reports to establish the recorded history of the TMI-2 condensate polisher for TMI-2, as summarized and reported in Appendix H.

It is noted that the condensate polisher was one of the equipments at TMI-2 with the poorest maintenance history (Appendix H). (The status report of a post accident inspection indicates a lack of preventative maintenance.) At the time of the March 28, 1979, event, there were 13 open work requests against the condensate polisher. Most of them concerned leaks and malfunctioning instrumentation. In addition, 13 work requests had been worked on during the 3 months prior to the event. These problems also were associated with leaks and malfunctioning instrumentation. It is obvious from this maintenance history that the problems with this piece of equipment were excessive and that there are possible design problems. It was also noted that these work orders were not covered by quality control, and that inspections were not performed as the work was accomplished, or after it was completed, to verify proper accomplishment of the work and to verify that unauthorized work or disturbances were not done (Appendix R).

The only history of problems was in the work requests file. This is discussed in general terms in the above paragraph. The specific work requests are listed in summary form in an enclosure to Appendix H. The condensate polisher is not listed as safety-grade or "safety-related" equipment; this may explain why reports related to its condition or performance are not in the other discrepancy files. For example, no LERs on the condensate polisher were found. No trend information has been located. That the condensate polisher was not classified as safety related meant: that this equipment was not covered by inspection (quality control), that changes to its procedures were not reviewed by independent reviewers and management review groups, and that it did not receive attention from management, despite the fact that its function was vital to the continued reliable operation of the plant.

Findings

- The condensate polisher was not covered by quality control during operation.
- No systematic evaluation of its "health" was accomplished.
- There is a history of leaks and malfunctioning instrumentation related to the condensate polisher.
- The number of work requests to fix the condensate polisher was large when compared to the history of other equipment.
- Thirteen work requests were open at the time of the TMI-2 accident; none were related to polisher 7.
- No problems on the condensate polisher, which was not "safety-related," were reported on LERs; none were required.
- Management and management review groups were not aware of the "health" of the condensate polisher, despite its being vital to plant operation.

RELATION TO ANOTHER EQUIPMENT PROBLEM

Another possible clue to what caused the polisher effluent valves to close during the March 28, 1979, accident is noted in reference 2, where it was noted that the emergency feedwater control valves appeared not to respond normally, as though they had lost instrument air supply during the early part of the accident. Both the polisher effluent valves and the emergency feedwater valves are reported to depend upon the instrument air supply operation. This suggests a possible loss of instrument air supply to both actuating devices at approximately the same time; perhaps a single point failure exists that has been unrecognized.

Findings

- The instrument air supply may have been lost to more than one activity at the beginning of the accident.

POST-ACCIDENT INVESTIGATION OF CONDENSATE POLISHER

As summarized in Appendix I, tests have been performed on the TMI-2 condensate polisher in an attempt to discover the reason why the effluent valves went to the closed position on March 28, 1979. The tests have been done with only one polisher, with partial simulation of the conditions that existed on March 28, and with water in the instrument air line. The effluent valves fluttered but did not close.

Subsequent to these tests, while replacing one polisher with another on July 5, the effluent valve of the one polisher in use was observed to flutter and then close. After removing water from the air supply line, the polisher was placed back in service and performed satisfactorily; this event is in the enclosure to Appendix I.

A systems analysis has been performed and reported in Appendix P that postulates a way in which water accumulating in control air line, from the water sluicing used to remove resin, could trigger a series of events that possibly could result in reproducing the closure of effluent valves on March 28. Key in the analysis is close reproduction of the flow conditions, valve positions, and water in the control air line as they were on March 28.

Findings

- Tests to date at TMI-2 have not confirmed a reason for closure of the effluent valves on March 28.

- Effluent valve on the one polisher in operation on July 5, 1979, went to closed position when polishers were being exchanged; water was found in the air line.

- A systems analysis postulates one way in which water in the air lines could possibly cause effluent valves to close as on March 28.

DETAILED FINDINGS

The individual findings, in order of their appearance in the text, are listed below for convenience. They have been consolidated into significant findings in the following section of this report.

- The condensate polisher effluent valves closed at the beginning of the accident on March 28, 1979.
- Water was found in the air systems.
- Polisher bypass valve failed to open.
- The condensate polisher effluent valves had closed unexpectedly in 1977 and 1978 tests, under similar surveillance procedure conditions.
- Design deficiency in the bypass valve was not corrected.
- Problems in removing spent resin remain unresolved.
- Problems were not thoroughly investigated and analyzed.
- Continued problems were experienced with water in the instrument and service airlines.
- Recommended corrective actions were not thoroughly considered.
- Corrective actions were not verified before use.
- The condensate polisher was not classified as "safety-related" equipment; thus it did not require design analyses and tests as in the QA program.
- The condensate polisher as used at TMI-2 has essentially a zero operational margin.
- Use of an appropriately designed bypass valve could have provided a substantial operational margin.
- The condensate polisher bypass valve was not designed to automatically actuate; nor was it designed for emergency manual actuation although apparently considered so in the TMI-2 FSAR.
- There were problems with resin removal from the polisher tank; problems developed about once each 12 transfers.
- Design changes were not always coordinated through the designer after 1971.
- Available system drawings were significantly different from equipment. No interface drawings nor integrated system schematics were available.

- Control of surveillance procedure used on polisher was lax.

- A recently revised surveillance procedure, not verified before being put into use, may have been in use on March 28, 1979.

- There were difficulties encountered with polisher 7 during the surveillance procedures before March 28, 1979, and again on March 28.

- The condensate polisher was not covered by quality control during operation.

- No systematic evaluation of its "health" was accomplished.

- There is a history of leaks and malfunctioning instrumentation related to the condensate polisher.

- The number of work requests to fix the condensate polisher is large when compared to the history of other equipment.

- Thirteen work requests were open at the time of TMI-2 accident; none related to polisher 7.

- No problems with condensate polisher, which was not classified as "safety-related," were reported on LERs; none were required to be reported.

- Management and management review groups were not aware of the "health" of the condensate polisher, despite it being vital to plant operation.

- The instrumentation air may have been lost to more than one activity at the beginning of the accident.

- Tests to date, at TMI-2, have not confirmed a reason for closure of the effluent valves on March 28, 1979.

- Effluent valve on the one polisher in operation on July 5, 1979, went to a closed position when polishers were being exchanged; water was found in the air line.

- A system analysis postulates one way in which water in the air line could possibly cause effluent valves to close as on March 28, 1979.

SIGNIFICANT FINDINGS AND CONCLUSION

Studies, analyses, inspections and tests of the Three Mile Island TMI-2, condensate polisher have resulted in the following significant findings and conclusion.

1. The condensate polisher effluent valves closed at the beginning of the accident on March 28, 1979.

2. These valves had closed unexpectedly twice before under similar surveillance procedures, and again later under a related condition.

3. Tests at TMI-2, to date, have not confirmed a reason for the closure. A possible reason has been postulated by an analysis which shows a way that water accumulating in the control air line could trigger valve closure.

4. The condensate polisher as used at TMI-2 had essentially zero operational margin.

5. The polisher bypass valve is not designed for automatic emergency opening nor even for manual opening in an emergency, although its availability is noted in the FSAR. If appropriately designed, this valve could provide substantial operational margin.

6. The condensate polisher valve failed to open by remote control during the accident, and at least once before; and corrective action had not been accomplished.

7. Resin removal from polisher has been a chronic problem, and has existed since early system tests. Corrective action has been ineffective. Removal problems appear to contribute to unexpected closing of effluent valves. Design problems were not worked out with the polisher designer.

8. Available drawings at TMI are significantly different from the condensate polisher; no interface drawings nor integrated schematics are available for use by procedure writers, maintenance people, or system engineers.

9. Condensate polisher is not classified as "safety-related," did not receive detailed design analyses, did not receive quality control coverage during operation, and did not receive management and management review group attention for problems with hardware or for procedure control.

10. There is a history of operational problems and a large amount of maintenance work on the polisher.

CONCLUSION

The condensate polisher, although vital to the operation of the power plant, did not receive appropriate attention in design and from assurance function, engineering, management, and management review groups; proper attention could have provided a significant increase in plant **reliability**.

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These documents are part of the Commission's permanent records that will be available in the National Archives.

REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

CLOSED EMERGENCY FEEDWATER VALVES

BY

William M. Bland, Jr.

October 1979
Washington, D.C.

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SUMMARY

On March 28, 1979, the Three Mile Island (TMI-2) nuclear power plant experienced the most severe accident in U.S. commercial nuclear power plant operating history. The accident, which occurred at about 4:00 a.m., was started by a loss of normal feedwater supply to the steam generators, which led rapidly through a normal sequence of events for reactor shutdown. During this normal sequence of events, the pilot-operated relief valve (PORV) opened and stuck (failed) in the open position where it remained for a considerable length of time before being isolated by action of the operators. This failure, followed by early operator action that throttled the flow from the high pressure injection (HPI) pumps, initiated an abnormal sequence of events that led to this most severe accident.

A number of analyses, involving studies, inspections, and tests have been conducted to understand what caused this accident and why it happened. One of these analyses was to investigate the reason for the emergency feedwater (EF) valves being in the <u>closed</u> position instead of the <u>open</u> position as required, as described in this report.

The findings and conclusions from this analysis are as follows.

<u>FINDINGS</u>

1. There has been no positive identification of a reason for the valves being in the <u>closed</u> position.

2. Of all the explanations analyzed, the most likely explanations, each with comments to the contrary, are:

- The valves were not reopened at the conclusion of the most recent surveillance procedure, requiring them to be <u>closed,</u> conducted prior to the accident.
- The valves may have been mistakenly <u>closed</u> by control room operators during the very first part of the accident.
- The valves may have been mistakenly <u>closed</u> from other control points within the plant.
- While considered a remote possibility, there is a chance that these valves were <u>closed</u> by an overt act.

3. A number of deficiencies have been identified during this analysis to determine why the emergency feedwater valves were in the <u>closed</u> position. These deficiencies are highlighted by the following findings:

- Nuclear safety-related procedure change requests to close these valves during surveillance testing did not receive proper technical evaluation. The Nuclear Regulatory Commission (NRC) failed to detect this violation of technical specifications during inspections August 1978 through March 26, 1979.

- The as-run checklist of surveillance procedure involving emergency feedwater valves was not reviewed or retained.

- Verification of important procedural steps, as by inspection, was not accomplished and recorded, nor was it required.

- There was no periodic systematic review of control room status.

- Too many respondees were used during checklist call out.

- Too many people had access to potentially sensitive plant locations.

- Switches and valves do get mispositioned, possibly more frequently than formal records indicate.

- The TMI-2 emergency feedwater valves had a history of only a few problems.

- Two conditions that were not to drawings and specifications were found in the emergency feedwater circuitry despite being under Quality Assurance Program control, as required by safety-related classification of these valves; these conditions did not affect operation of the valves.

4. There is physical evidence that at an unknown time an unexpected event or transient caused overheating in the emergency feedwater system. It is likely that the cause of the observed condition occurred after the emergency feedwater valves were <u>opened</u> by operator action and played no part in the reason why the valves were in the <u>closed</u> position.

5. Tests and inspections, in place of a sophisticated sneak circuit analysis, did not find a sneak circuit path that would operate the emergency feedwater valves.

6. Emergency feedwater valve position indicator circuitry has been confirmed to be in working order.

<u>CONCLUSIONS</u>

1. The utility failed to apply appropriate control over safety-related procedure and its implementation and changes to it; NRC failed to detect lack of control.

2. The utility does not apply appropriate discipline to access to in-plant areas, accomplishment of procedures, and equipment configuration. NRC did not recognize this lack of discipline.

INTRODUCTION

On March 28, 1979, TMI-2 of the nuclear power plant at Three Mile Island (TMI) near Middletown, Pa., experienced the most severe accident in the history of commercial nuclear power plant operations in the United States. This accident has been, and continues to be, the subject of much investigation in order to determine the primary cause behind the initiation and furtherance of the accident.

A number of analyses, involving studies, inspections, and tests have been conducted to understand what caused this accident and why it happened. One of these analyses, which investigates the reason for the emergency feedwater valves being in the closed position instead of in the open position as required, is presented in this report.

The importance of these valves is indicated by their classification as safety-related and the fact that it is noted by procedure and in the technical specifications that they are to be in the open position during plant operation.

Being in the closed position at the beginning of the accident prevented the emergency feedwater from reaching the steam generators in the first minute after loss of the normal feedwater. At a little over one minute (reference 1), into the accident, the operators noted that the steam generators had "gone dry" which resulted in loss of the capability of heat transfer from the reactor coolant system to the secondary system (reference 8). At about 8 minutes into the accident (reference 1), the operators "positioned" these valves to the open position and quickly re-established feedwater flow to the steam generators.

It is unlikely that the delay in establishing emergency feedwater flow directly affected the course of the accident; however, it did have an intangible effect in that it did provide a significant distraction to the control room operators who were already busy with emergency conditions.

ANALYSIS PLANS

This report, which is really more in the nature of an investigative than an analysis report, has been prepared to present the information that has been learned about why the emergency feedwater valves were in the closed position at the time that the TMI-2 accident began and not in the open position as required by specification and procedure. The intent is to document the results of this investigation and related findings which will be useful in preventing a recurrence.

Much work in the area of the closed emergency feedwater valves, sometimes identified as EF-V-12A and EF-V-12B, has already been accomplished and reported by others (references 1 and 2). It is the intent not to redo this earlier work, but to make maximum use of it where it is in agreement with this analysis. Details in this report will discuss where differences occur or where additional investigations were or are being conducted as part of this present analysis.

GENERAL DESCRIPTION OF FEEDWATER SYSTEM

Details of the feedwater system, including the emergency feedwater system, can be obtained from the text and schematic in reference 3. Some general information about this system from reference 3 is contained in the following paragraphs.

The condensate and feedwater system is designed to supply feedwater to the steam generators from the condensate produced in the condenser during normal power operation. System design is based on calculated heat balance and two parallel condensate and feedwater trains are provided.

The system is also designed to supply feedwater to the steam generators and to maintain an emergency high water level in the steam generators in the event of loss of both main feedwater supply trains.

There are also available two motor driven emergency steam generator feedpumps and one turbine driven emergency steam generator feedpump. These pumps are fed from any of the following sources: the condensate pump discharge, the condensate storage tanks, and either redundant branch of the nuclear services river water system.

ANALYSIS

Much work in the area of the closed emergency feedwater valves, EF-V-12A and EF-V-12B, has been acco lished and reported in referenced documents. To make best use of this earlier work, material from reference 1 will be quoted in this report and it will be noted where analyses are in agreement. Where the present analyses have differences and where different or additional investigations were (or are being) conducted, the differences will be discussed. In this way, the present state of knowledge can be gathered in one place and assessed more readily.

MATERIAL FROM REFERENCE 1

The following possible reasons for the emergency feedwater valves having been in the closed position at the time the TMI-2 accident began are noted below, as transferred from reference 1.

1. The valves were left closed after the last surveillance test of the emergency feedwater system.

2. The valves were closed by the overt act of an individual.

3. The valves were left closed after maintenance work on the system.

4. The valves malfunctioned as a result of an improper design change or plant modification.

5. The valves malfunctioned because they were exposed to elevated temperatures prior to or during the accident.

6. The valves were closed as an operator action prior to or during the transient.

Review of all these possible causes revealed no reason to believe that any of them was the specific cause of the closed valves. The findings are summarized below:

1. The operators and supervisors responsible for conducting the surveillance test on March 26, 1979, were interviewed. . . . The operator who actually manipulated the valves involved stated that he specifically recalled opening that valve. The investigation found no basis for rejecting his assertion. If his assertion was incorrect and the valves were left closed after the test, the investigation found no information to explain how the closed valves would have gone unnoticed during the 42 hours between the test and the accident. However, routine panel inspections are not required of the staff by this licensee.

2. No information was developed during this investigation indicating that sabotage was a contributing factor to the initiation of the accident or to the subsequent response of plant personnel or equipment to the accident.

3. The possibility of maintenance work being done on the valves was addressed. . . . No evidence was found of such maintenance after both record reviews and interviews.

4. The possibility that the valves were closed as the result of an improper response of the valve control circuits to the turbine trip was addressed. A change was made in the logic circuity related to the operation of the emergency feedwater valves. The change included defeating the automatic closure of the emergency valves EF-V-12A/B with a low once-through steam generator (OTSG) pressure signal. This feature had been part of the protective circuitry involved with the plant response to a steam break accident. If the modification required by 9.1 had not been properly performed, there would be a possibility of the valves closing. Since the accident, the licensee has written and performed a test to determine if the closure demand feature had been removed from the EF-V-12A/B valves. The results indicate that the valves did not close when the feedwater latching logic was introduced, indicating that the changes affecting EF-V-12A/B appear to be correct. Moreover, the pressure in the OTSGs during the first 8 minutes did not reach the initiation point for this control system, even if the change had not been properly completed.

5. The possibility that the valves were closed, as a result of temperature problems as might occur from system backflow, was addressed. Information was obtained that suggests at least one of the valves might have undergone a thermal transient. This was based on observed discoloration of the valve piping. The visual inspection by an investigator confirmed that a plastic instruction tag on valve EF-V-11B, the EFW control valve, was "melted." The investigation included a review of possible reverse flow paths to the loop B OTSG, a check of maintenance requests, and interviews with mechanical and electrical maintenance personnel and operations personnel. Burns and Roe drawing no. 2005, flow diagram feedwater and condensate, shows the possible flow paths from B OTSG. A backflow from inside containment would have to travel through reactor building penetration R-616B, check valve. An alternative path could involve the same penetration, EF-V-13B, EF-V-12B, EF-V-32B, and end at the backside of EF-V-11B on to EF-V-12B. A third path could include the penetration R-616B, EF-V-13B, EF-V-32B, and back up to EF-V-11B and/or through EF-V-32B. The discoloration of the pipe appears to indicate heating along the pipe from penetration R-616B to the check valve EF-V-13B through EF-V-12B to EF-V-11B, the most direct route.

   The possibility that oil staining might indicate an overheating of these valves was addressed. The EF-V-12B valve appears to have oil leakage from the limitorque operator motor which stained the valve body and piping. No evidence of a work request for the EF-V-12 valve just prior to March 28, 1979, was found. Operations auxiliary operators who performed the EF surveillance test that required them to be in the vicinity ur the EF-V-12B valve were interviewed regarding the valve oil leak. Five stated they did not recall seeing an oil stain, while the sixth did not recall looking at that valve. They did perform the surveillance over a period from January 3, 1979, to March 3, 1979. Additional information presented to the investigator indicated that the valve EF-V-12B did not have an oil stain on March 26, 1979. The investigator did note that the instruction tag on EF-V-11B was deformed and showed signs of being burnt (brown) on the rear side where it is in contact with the valve housing.

The condition of the EF-V-12A valve and piping was inspected and found sound with no similar condition. On March 28, 1979, both EF-V-12A and EF-V-12B were in a closed status.

There was no evidence to cause the investigator to conclude that either EF-V-12 would be closed because of the condition of EF-V-12B or the condition of the B emergency feedwater piping. All information indicates that both valves opened when actuated by the control room operator on March 28, 1979, at about 4:08 a.m. This review did not conclude how the emergency feedwater pipe became discolored, how the oil leaked, nor how the tag deformed. The purpose of this study was to determine if the condition could have been a reason for the EF-V-12B valve to be in a closed position at 4:00 a.m. on March 28, 1979. The findings do not indicate a relationship. The possibility of a correlation to the status of the B OSTG emergency feedwater piping after its isolation during the accident was not pursued.

6. The possibility that the valves were closed as an operator action during the transient was addressed. The operating staff on duty during the period when the valves were found closed were interviewed to determine whether these valves could have been closed as an operator action to prevent an excessive cooldown rate of the reactor coolant system (RCS) and an attendant pressurizer level drop. The investigators pursued the possibility that the action was initially taken and then forgotten by the operator for 8 minutes. No information was obtained during this interview that would indicate that this operator action took place during the accident.

ANALYSES IN CONJUNCTION WITH THOSE OF REFERENCE 1

The results of the investigation conducted by the technical staff agrees to a large degree with the foregoing results of reference 1 in the area of investigation to determine why the emergency feedwater valves were in the closed position when they were required to be in the open position. A point-by-point comparison follows:

1. <u>The valves were left closed after the last surveillance test of the emergency feedwater system.</u>

This analysis agrees with the analysis of reference 1, but adds that failure of the operators to retain the check sheet used to mark their progress through the accomplishment of Surveillance Procedure 2303-M27A/B on March 26, 1979, as noted in reference 4, and the description where two control room operators were responding to the surveillance team in reportedly positioning the emergency feedwater valves by actuation of valve controls on the panel in the control room to the open position at the completion of the surveillance procedure on March 26, noted in Appendix B, leave doubt that the valves were opened as reported.

A quote from reference 6 reinforces the doubt related to this point:

> The CR0 (control room operator) assigned in relief shift during surveillance testing 3/26/79 on EFW system stated that he remembers Auxiliary Operator reading off valves to be realigned at the completion of the test, but he does not remember whether he performed the operation to attain the EF-V-12 valves or whether the CR0 on shift performs the operation. He stated they were both standing at the board and both responding alternately, apparently, although he was not positive in the point. This confirms the report given by the Auxiliary Operator.

During this review it was noted that the accomplishment of Surveillance Procedure 2303-M27A/B, revision 4 on March 26, 1979, was the last time the emergency feedwater valves had been operated before the March 28, 1979, accident. If the feedwater valves had been left in the closed position then indicator lights on the panel would have so indicated that position. This implies several shift changes occurred without notice of wrong position of valves. Such an error would normally be picked up, at least, during a well disciplined shift change.

It was noted in reference 6 that no inspector witnessed the accomplishment of this surveillance procedure, as is done for important procedure accomplishment in other high-risk activities, even though this system is classified as safety grade. The responsibilities for such inspection coverage are left up to the Quality Assurance (QA) Department by Procedure 1001, Three Mile Island Nuclear Station, Station Administration Procedure 1001, Document Control, Appendix M, as quoted from paragraph 2.4, as follows:

> The Quality Assurance Department has the option to survey any and all procedures. Those procedures chosen for QA [quality assurance] surveillance will be indicated with the words, "QC Hold Points Indicated" or "Performance to be observed by Quality Control. Notify QC at least four hours prior to starting task," on the cover sheet....

From interviews with the quality control (QC) manager for TMI, it was learned that his staff is able only to survey (or inspect) an average of about one application of each surveillance procedure every 2 years

(Appendix N). Thus there is no real requirement to accomplish inspection of important procedures on a regular basis. There is no NRC requirement for the inspection to be performed.

It also was noted that a Procedure Change Request, No. 2-78-707 (Appendix A), applicable to this procedure, had been initiated on August 1, 1979, had been approved on August 15, 1978, and had resulted in the issuance of the procedure on August 30, 1978, as revision 4. The reason for the change request was given as:

> New Pump reference valves established because valve line up is changed. EF-V-12A/B are now closed because EF-V11A/B was [sic] leaking by. With EF-V12A/B closed, old flow rate cannot be duplicated.

Thus the change in this surveillance procedure results in planned closure of both emergency feedwater valves at the same time each time the procedure was applied (a similar change was made to a related procedure, but it is not considered necessary to evaluate both in this analysis).

The change request (Appendix A), classified as a nuclear safety-related change, was processed by procedure, but as noted in Reference 1, the safety evaluation failed to address the aspect of the change which called for simultaneous closing of both the emergency feedwater valves at the same time and thus isolating the emergency feedwater pumps from the steam generators. The closure of these valves appears to impact the TMI-2 technical specification. For this kind of impact, Metropolitan Edison Company (Met Ed) should have processed the change through NRC <u>before</u> making the change to the procedure, thus Change Request Procedure AP 1001 was not followed. This finding is noted in reference 1 and in reference 6, the post-accident review conducted by Met Ed/General Public Utilities Service Corporation (GPUSC).

It also is noted that NRC did not detect this violation of the technical specification despite frequent Office of Inspection and Enforcement (I&E) visits to TMI-2, as evidenced by a summary of inspection reports (Appendix C), which indicates 15 inspection periods with indication of few noncompliances found between August 1978 and March 1979. In addition, it is noted in the report (Appendix D) that an I&E inspection was made at TMI-1 and TMI-2 March 19-23 and March 26, 1979. Surveillance Procedure 2303-M27A/B was last accomplished before the accident on March 26, 1979.

Findings from this analysis are:

- A nuclear safety-related procedure change request did not receive proper technical evaluation.

- NRC failed to detect this violation of test specification during inspections August 1978 through March 26, 1979.

- As-run checklists of important procedure are not reviewed and signed by appropriate levels of supervision and are not retained to support what was accomplished.

- Verification of actual accomplishment of significant steps in important procedures was not accomplished and recorded by an unbiased party (as by an inspector), nor was it required.

- There was no periodic systematic review of control room equipment status to assure that it meets operational requirements.

- Too many operators were responding to checklist call out in final positioning of the emergency feedwater valves at the completion of Surveillance Procedure 2303-M27A/B on March 26, 1979.

2. <u>The valves were closed by the overt act of an individual.</u>

This analysis closely agrees with the finding of reference 1, particularly since it seems unlikely that a knowledgeable individual intending to cause damage would select the relative insignificance of these valves as their target. However, a person not very knowledgeable may have made a mistake. Also, from information contained in the analysis in reference 1, hundreds of people had access to these positions from which the valves could have been controlled. Information supplied by Met Ed during this investigation (Appendix H), indicates that as many as 728 people on March 26, 758 on March 27, and 81 during the first 4 hours of March 28, had access to locations in the plant from which they could have controlled the position of these valves.

The fact that so many people had access to the three control points for these valves has made it necessary for a request to the Federal Bureau of Investigation (FBI) to consider performing an investigation (Appendix E) just to confirm that this is not a reason for the valves being in the closed position. The FBI response (Appendix E-1) was that they did not consider information and concerns expressed in Appendix E sufficient cause to initiate an investigation.

In addition to the analysis reported in reference 1 and to the request of Appendix E, this technical effort also analyzed personnel records selected by Met Ed through the screening process noted in Appendix F. Of the five records produced by Met Ed, one was related to a person verified by Met Ed as being absent from the TMI-2 site prior to and up to the time of the accident and another was related to a person identified by Met Ed as requiring an escort in that part of TMI-2 where the valve controls are located (Appendix F-1). No other investigation has been conducted nor is any planned of the remaining three records (Appendix F-1).

While the results on hand are not conclusive, it appears that there is only a remote possibility that the reason for the valves being in the <u>closed</u> position is because of an overt act.

Findings from this analysis are:

- The policy and implementing procedures for controlling access to various parts of the nuclear power generation plant permitted access of too many people to many potentially sensitive locations.

° While considered a remote possibility, there is a chance that these valves were closed by an overt act.

3. The valves were left closed after maintenance work on the system.

This analysis has no information contrary to that of reference 1. This is supported by a review of work requests; none was found related to the valves in the period March 25, 1979, thru March 28, 1979. Only two work requests related to these valves were found in the period November 1978 to August 1979. One was completed July 10, 1979, and the other is still open. It is concluded that no work was done on these valves after the March 26, 1979, surveillance procedure had been accomplished. From the results on hand it appears unlikely that this is the reason for the valves being in the closed position.

4. The valves malfunctioned as a result of an improper design, change or plant modification.

This analysis has no information contrary to that of reference 1 at this time. Sneak circuit inspection and tests conducted on the circuits related to these valves (see next section) by request of this analysis team did not provide additional information.

From the results on hand it appears unlikely that this is the reason for the valves being in the closed position.

5. The valves malfunctioned because they were exposed to elevated temperatures prior to or during the accident.

This analysis has no information contrary to that of reference 1 at this time. An engineering evaluation of the reported visual evidence that an unexpected event or transient occurred in the emergency feedwater system related to EF-V-12B is being accomplished. The time of the event or transient is unknown. Evaluation results so far merely confirm that an overheating condition did occur, and speculate that the overheating may have come from backflow, after the EF-V-12A/B valves were opened by operator action and at a time when there was a pressure difference between steam generator A and B (Appendix K). Investigation into possible paths of this flow is noted in Appendix L. Status reports, Appendix P, further postulate that hot water from steam generator A backed into the B system lines, when steam generator A pressure was more than 50 psi higher than the pressure in steam generator B (after about 1.5 hours into the event) and assuming a failed check valve in B line. Tests correlate damage to paint with water temperature available.

From the results on hand and because the time span from accident initiation to the time these valves were discovered to be in the closed position, only 5 minutes (reference 1), is so short it is doubtful that a thermal transient from inside the steam circuitry could have had an adverse effect and forced the valves to the closed position and because the valves opened when commanded to do so indicates no significant damage of a thermal transient had occurred then. It appears unlikely that the observed conditions of discoloration of pipe, stains on valve, and deformation of plastic tag attached to the valve will provide a reason for concluding the valves were in the closed position.

The findings from this analysis are:

- An unexpected event or occurrence or flow path caused the observed conditions of discoloration and stains on the pipe and valve in the emergency feedwater system and deformation of an attached plastic tag.

- It is likely that the cause of the observed condition occurred after the emergency feedwater valves were opened by the operators.

6. The valves were closed as a result of an operator action prior to or during the transient.

This analysis has no information contrary to that of reference 1 at this time. However, it does seem possible that such an operator action could have taken place in the excitment of the very early stage of the accident at one of the valve control points.

From the results on hand, it appears that it is possible, though remotely so, that an operator action at one of the valve control points in the plant could have been the reason for the valves being in the closed position.

ADDITIONAL ANALYSES

1. Analysis of the documentation formed during the inquiry accomplished at the request of GPUSC by John Miller, GPUSC executive consultant, into why the emergency feedwater valves were found in the closed position instead of in the required open position.

2. Analysis to determine if a sneak circuit had closed the emergency feedwater valves.

3. Analysis of records search to determine how prevalent the problem is of having valves and switches found in wrong position.

4. Analysis of results of an inspection of position indicator switches and circuitry on the emergency feedwater valves.

The results of investigation and analysis of these areas follows:

1. The documentation (Appendix B) of the inquiry into why the emergency feedwater valves were in the closed position was made by John Miller, GPUSC executive consultant, and E. O'Connor of Jersey Central Power & Light Company. The inquiry was conducted in considerable detail with almost all logical questions asked. The inquiry confirmed that the existing documentation indicates to a limited degree that the valves were thought to have been opened by the operators at the completion of the surveillance test conducted on March 26, that no other tests

involving these valves were scheduled before the March 28 accident and that it was unlikely that valves were commanded to the closed position from any of the three control positions in the plant.

Individual depositions were taken from O'Connor and Miller. O'Connor concluded he did not know how the valves reached the closed position (reference 5).

Miller stated during his deposition that he believed the best possibility of an explanation is that the valves were not reopened following the surveillance test on March 26 and believes that is the best that will ever be known on this subject (reference 6). Miller appeared vague during the deposition regarding toward whom he addressed questions during his inquiry; that is, he tended not to recall the names of individuals he interviewed, as he conducted his inquiry. This, to some extent, tends to raise a question as to the vigor applied during his search for the reason the valves were in the closed position.

The investigation conducted by Miller and O'Connor appears to have been fairly complete. Only one person appears to have been missed by their interviews of the persons closest to these valves during the surveillance test -- Cooper. Cooper was reported to have been away from the TMI site during the period of Miller's inquiry because of illness; however, Cooper later appeared before the Commission on May 20, 1979, with other persons who conducted these tests and testified that he did open these valves (at the conclusion of the surveillance test).

An analysis of this inquiry brought forth no new explanation, the first part of this analysis, and reinforces doubt that the valves were opened as reported at the conclusion of the Surveillance Procedure involving the emergency feedwater valves completed on March 26, 1979.

The finding from this analysis is that there is a possibility that the emergency feedwater valves were not reopened following the March 26, 1979, surveillance test.

2. Another investigative step was initiated. That is, to search by inspection and test means (wherever possible) for a "sneak circuit" that could have electrically commanded these valves to the closed position by some unique combination of conditions, switch positions, equipment failure, or equipment operation that occurred in the early phase of the March 28 accident. Results of a sophisticated sneak circuit analysis performed on a reactor that had been in service a number of years (reference 7) has been forwarded to GPUSC Systems Engineering Organization. This organization initiated a more simple search for sneak circuit associated with these valves. The search involved hands-on inspection of the involved circuity coupled with simple electrical measurements in an effort to identify unexpected electrical paths.

This search that involved tests and inspections of the electrical circuitry related to the emergency feedwater valves in an effort to identify a sneak circuit condition that would cause these valves to be commanded to the closed position has been completed by GPUSC engineers. They report, as documented in Appendix I and Appendix 0, "there were no unexpected circuits found, nothing found to indicate a sneak circuit condition that would explain why the EF-V12A/B valves were found in the closed position...." It is emphasized that the test and inspection carried out is a physical check and, because of configuration and geometry limitations, may not provide the assurance that a sneak circuit did not exist as a sophisticated sneak circuit analysis (reference 7) would do.

These tests and inspections did uncover two conditions in these circuits which were not done according to drawings and specifications despite these circuits being under QA Program control (Appendix I); neither condition is suspected of causing the valves to be in the closed position.

The findings from this investigative step are as follows:

- Tests and inspections did not find a sneak circuit path that would operate the emergency feedwater valves.

- Two conditions were found in these circuits that were not to drawings and specifications.

3. A search of available records has been made to determine the frequency with which valves and switches are found in other than expected positions at TMI, and the history of problems with the emergency feedwater valves.

The TMI-2 nonconformance records, including the licensee event reports (LER), were searched and there was no record of either switch or valve being reported in the wrong position (Appendix J).

The TMI-1 records were similarly searched, extending back to Jan. 25, 1973. The records showed seven instances (Appendix J) where switch(es) and/or valve(s) had been found in the wrong position. This was about one reported instance per year. In each case the report was based upon some incorrect function or a failure to function. It is therefore assumed that the failure was the reason for the report and not the wrong position.

Early informal interviews with TMI-2 personnel resulted in the explanation that occurrences of switches and valves found in the wrong position would be worked out and fixed with the shift supervisor at once without a formal report unless a failure to function had been the reason for the detection.

All available nonconformance and deficiency records at THI and the LER system were searched for reports of problems with emergency feedwater valves at TMI-2 (Appendix G). Only two reports, in the maintenance historical records, were written on these valves; neither can be related to these valves being found in the closed position on March 28, 1979.

No reports were found in the LER system on emergency feedwater valves in Babcock & Wilcox Company (B&W) plants.

From this analysis the findings are:

- Switches and valves do get mispositioned, possibly more often than formal documents indicate.

- Emergency feedwater valves EF-V-12 have little history of problems.

4. Sometimes switches can malfunction or get out of adjustment to provide misleading position indication, as noted in reference 1, for the EF-V-11A/B valve switch and indicator. During the tests and inspections conducted for possible sneak circuit electrical paths, the valve position indicator switch and circuity for each emergency feedwater valve, EF-V-12A/B, were checked and confirmed to be in proper working order (Appendix I and Appendix 0).

The finding from this analysis is that emergency feedwater valve position indicator circuity was confirmed to be in working order after the accident.

FINDINGS AND CONCLUSIONS

Analyses conducted into determining the reason the emergency feedwater valves were in the closed position when needed early in the TMI-2 accident on March 28, 1979, instead of the open position, have resulted in these findings and conclusions.

FINDINGS

1. There has been no positive identification of a reason for these valves to have been in the closed position.

2. Of all the explanations analyzed, the most likely explanations, each with comments to the contrary, are:

   - The valves were not reopened at the conclusion of the most recent surveillance procedure, requiring them to be closed, conducted prior to the accident.

   - The valves may have been mistakenly closed by control room operators during the very first part of the accident.

   - The valves may have been mistakenly closed from other control points within the plant.

   - While it is considered a remote possibility, there is a chance that these valves were closed by an overt act.

3. A number of deficiencies have been identified during this analysis to determine why the emergency feedwater valves were in the closed position. These deficiencies are highlighted by the following findings:

   - A nuclear safety-related procedure change request to close these valves during surveillance testing did not receive proper technical evaluation. NRC failed to detect this violation of technical specification during inspections August 1978, through March 26, 1979.

   - The as-run checklist of surveillance procedure involving emergency feedwater valves was neither reviewed nor retained.

   - Verification of important procedural steps, as by inspection, was not accomplished and recorded; nor was it required.

   - There was no periodic systematic review of control room status.

   - Too many respondees were used during checklist call out.

- Too many people had access to potentially sensitive plant locations.

- Switches and valves do get mispositioned, possibly more frequently than formal records indicate.

- The TMI-2 emergency feedwater valves had a history of only a few problems.

- Two conditions that were not to drawings and specifications were found in the emergency feedwater circuitry despite being under QA Program control as required by safety-related classification of these valves; these conditions did not affect operation of the valves.

4. There is physical evidence at an unknown time that an unexpected event or transient caused overheating in the emergency feedwater system; it is likely that the cause of the observed condition occurred after the emergency feedwater valves were opened by operator action and played no part in the reason why the valves were in the closed position.

5. Tests and inspections, in place of a sophisticated sneak circuit analysis, did not find a sneak circuit path that would operate the emergency feedwater valves.

6. Emergency feedwater valve position indicator circuitry has been confirmed to be in working order.

CONCLUSIONS

1. The utility **failed to apply appropriate control over a safety-related** procedure, its implementation, and changes to it. NRC failed to detect the lack of control.

2. The utility does not apply appropriate discipline to access to in-plant areas, accomplishment of procedures, and equipment configuration. NRC did not recognize this lack of discipline.

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REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

PILOT-OPERATED RELIEF VALVE
DESIGN AND PERFORMANCE

BY

Frederick Forscher
William M. Bland, Jr.

October 1979
Washington, D.C.

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SUMMARY

On March 28, 1979, at 4:00 in the morning, a feedwater transient at TMI-2 initiated a small-break loss-of-coolant accident (LOCA). Thus started the most serious accident in U.S. nuclear power history. The small-break LOCA was not recognized for over 2 hours, while contaminated cooling water streamed through the breach in the primary pressure boundary.

The small-break LOCA was caused by the pilot-operated relief valve (PORV) on top of the pressurizer that opened properly, but did not close when expected (thus, failed in the open position). Because of this PORV malfunction, and the early operator action to throttle the flow from the high pressure injection (HPI) pumps, the core was partially uncovered and the first barrier to the release of fission products was breached.

From every point of view, the breaching of any of the three barriers -- fuel cladding, pressure boundary, containment -- constitutes an unacceptable risk to the health and safety of the public and the environment. Any system, component, or action preventing or mitigating this risk should be considered "safety-related." Unfortunately, the PORV is not classified as a safety-related component.

This report contains the results of an investigation that touches on many aspects of the PORV -- its design, development, manufacture, testing, and history of performance and maintenance. It looks at the total process up to the eventual malfunctioning of the PORV at TMI-2, and is based on interviews, reports, and documentation from all involved organizations.

FINDINGS

1. The PORV apparently failed in the open position at TMI-2 on March 28, 1979; TMI-2 operators had no positive indication of the open/close position of the PORV; the absence of this signal in the control room contributed to the confusion of the operators during the TMI-2 accident.

2. Failure of the PORV in the open position resulted in a small-break LOCA.

3. Existing procedures did not consider a stuck-open PORV as a small-break LOCA.

4. The PORV was not classed as a safety-related component of the reactor coolant system.

5. The non-safety-related classification of the PORV can be traced to the application of the "single failure" criterion.

6. There have been 11 failures of pressurized water reactor (PWR) PORVs failing in the open position before TMI-2 documented in this report. Nine of these were in Babcock & Wilcox (B&W)

plants, compared with five recognized in the Nuclear Regulatory Commission (NRC) summary (NUREG-0560). A number of different causes are listed for these failures.

7. Of the nine that failed in the open position in B&W plants, B&W was aware of seven. Of the 11 that failed in the open position, 8 were supplied by one supplier, which was aware of 4 that were contained in the NRC summary (NUREG-0560).

8. Failure analyses and corrective actions following these PORV failures were insufficient and ineffective.

9. The NRC has not highlighted PORV problems as an unresolved safety issue, as an abnormal occurrence, or as a generic problem.

10. Standards for PORV design, testing, and function performance are not available.

11. National reliability data systems are only in the early stages of development by both the Electrical Power Research Institute (EPRI) and the NRC.

CONCLUSION

The TMI-2 accident would probably not have progressed beyond a severe feedwater transient had the PORV been recognized and treated as a "safety-related" component.

INTRODUCTION

ROLE OF THE PORV IN THE TMI ACCIDENT

The accident sequence is described in an NRC report (reference 1) and in a staff report (reference 3). The plant experienced a total loss of feedwater, initiated by a loss of condensate flow with an almost simultaneous trip of the main turbine. Reactor coolant system (RCS) temperature and pressure increased. The PORV opened as designed at its setpoint of 2,255 psig. The reactor tripped when the RCS pressure trip setpoint of 2,350 psig was reached.

The PORV failed to close when its lower setpoint was reached about 10 seconds later. Reactor coolant inventory loss continued through the open PORV. Pressure decreased to a low point of 660 psig at 2 hours and 19 minutes, when the leaking PORV was diagnosed and blocked by the closing of the isolation valve.

This failure of the PORV to close when the pressure decreased, followed by early operator action that throttled the flow from the HPI pumps, initiated an abnormal sequence of events that led to this severe accident.

Opening of the PORV in response to a pressure transient was an expected event and is listed in the turbine trip procedures. Additionally, the fact that the valve might stick open was covered by Emergency Procedure 2202-1.5 on pressurizer system failure. However, the possibility of a stuck-open PORV as the cause of a loss-of-coolant accident was not recognized by the loss-of-reactor-coolant emergency procedure. A more detailed discussion of the PORV and emergency procedures is contained in the staff report, "Technical Assessment of Operating, Abnormal, and Emergency Procedures" (reference 2).

THE PORV: WHAT IS IT? WHERE IS IT?

The "P" in PORV stands for "pilot," "power," or "pressure." The pilot-operated relief valve (PORV) is also called a pilot-actuated relief valve (PARV), or sometimes an electromatic relief valve.

Relief valves are set to relieve reactor coolant pressure at a level below the setpoint of the ASME (American Society of Mechanical Engineers) spring-loaded code valve (safety valve). This prevents the lifting of the spring-loaded ASME code valve(s) and reduces the maintenance requirement to reseat them necessitating cold shutdown of the plant.

The ASME code does not permit a blocking valve between the code safety valves and the pressurizer. Their pressure relieving capacity, for postulated conditions, must be demonstrated by test and in surveillance. The PORV <u>is introduced for operational convenience</u> to prevent the actuation of the code valve(s). The pressure-relieving capacity of the PORV is not included in the safety analysis of the RCS.

At the time of construction of TMI-2, ASME code valves cost about $10,000 and PORVs about $5,000 each (not installed). Today's prices are about four times the 1970 price.

All B&W plants employ Dresser pilot-operated relief valves, except Davis-Besse 1, where a Crosby-Aston Pressurematic valve is used. The designs are very similar, using a pressure-actuated electric signal to move the plunger in a solenoid.

Westinghouse uses pneumatically actuated PORVs. All plants supplied by Combustion-Engineering use Dresser valves. PORVs at B&W plants, except at Davis-Besse, are Dresser model 31533 VX-30, size 2-1/2 x 4 inches, with relief capacity that ranges from 12.6 to 14.1 kg/sec (100,000 to 112,000 lb/hr). The Davis-Besse valve is a Crosby model HPV-ST, 2-1/2 x 4 inches, 14.1 kg/sec (112,000 lb/hr).

PORVs are designed to reseat when the solenoid is de-energized, the pilot valve closes, and the pressure in the chamber beneath the main valve disc is restored.

A typical Dresser PORV is shown in cross section in Figure 1. Valve and controls are shown in Figure 2. Figure 3 shows a typical arrangement of a PORV and two code valves on the pressurizer of B&W plants. Two designs of Westinghouse PORVs are shown in Figures 4 and 5, to illustrate the difference in actuating mechanisms (pneumatic). B&W's informal design policy does not allow air-operated (pneumatic) devices and instruments in the containment building. Westinghouse's experience with their PORVs is considerably better than B&W's.

LONG HISTORY OF PORV FAILURES

The history of the malfunctioning of the PORV is confused and incomplete because: (1) many incidents were not reported because the PORV is not considered safety-related, and (2) most incidents that were reported are divided between instrument failures, electric component failures, and mechanical failures, but not PORV failures.

However, <u>a stuck-open PORV is a small-break LOCA, regardless of why and how it fails to close.</u> Table 1 lists 11 stuck-open PORVs (prior to TMI-2) in PWRs, and is discussed in more detail further on in this report. Of the 11, 9 had B&W-furnished valves (Palisades by Combustion-Engineering (CE) and Beznau by Westinghouse). Of the nine events involving B&W equipment, eight had PORV valves designed and supplied by Dresser. Thus, of the 11 known cases of stuck-open PORVs, eight had valves from one supplier -- Dresser Industries. The causes attributed to failure are varied, some are related to the valve and others related to failures of electrical power to the valve.

Except for Beznau (Westinghouse) and Palisades (Combustion-Engineering), all reactors listed in Table 1 are of B&W design. It is not possible with existing records to establish a complete history of PORV failures. Only 5 of the 11 events listed above are recorded and discussed in NUREG-0560 (reference 4). It is noted in Attachment 2 that the PORV

TABLE 1: Stuck-Open **Pilot-Operated** Relief Valves in Pressurized Water Reactors

REACTOR	DATE	ASSIGNED CAUSE
Palisades **	9/71	Loss of power
Oconee-2 **	8/73	Wiring error
Oconee-2 **	11/73	Pilot leakage
Beznau	8/74	Fractured housing
ANO-1 ~~**	8/74	Pilot vent line
Oconee-3 #**	6/75	Corroding leakage
Crystal River **	11/75	Stuck solenoid
Davis-Besse #	9/77	Missing relay
Davis-Besse	10/77	Pilot stem clearance
TMI-2 #**	3/78	Loss of power
Rancho Seco ~~**	6/78	Leakage
TMI-2**	3/79	Unknown

** Dresser Supplied PORV.

Reported in NUREG-0560.

Note: The Rancho Seco event of June 1978 was included in the open PORV listing of NUREG-0560, but was apparently of leakage variety.

supplier was only aware of the PORV problems contained in NUREG-0560 and not the others.

DESIGN REVIEW, STANDARDS, AND VERFICATION

Design-basis accidents, such as a postulated LOCA, form the boundary on the freedom of design. Valid codes and standards are further restrictions on the degree of freedom of designers. The ASME Boiler and Pressure Vessel Code, Section III, is such a restriction. It is referred to the world over as "the ASME code" and has served the interest of public safety for over 60 years, with or without supportive regulatory agencies. The NRC and other regulatory jurisdictions adopted the ASME standards and made it a statutory requirement; hence the name Boiler Code.

The PORV has never been subjected to similar conditions. Appropriate analysis depends, to a large extent, on valid standards that do not yet exist. It also depends on experimental verification of the functional performance of PORVs. Here, we lack the appropriate testing facilities and the confirmatory research that precedes such testing.

Functional (operational) deficiencies of relief valves have been of concern to professionals in industry and government for some time. Because valves play a major role in safety-related functions in nuclear power plants, nuclear code committees have tried for over 10 years to arrive at acceptable design and performance standards that can be verified (audited and tested) by independent third parties.

Our national standard effort has not yet resulted in usable standards, such as are referenced in appropriate regulatory guides. For example, proposed ANSI-B 16.41, "Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants" (Appendix B), has been under development for 7 years and is still far from becoming an ANSI standard. It would be appropriate for PORVs. There is no way for quality assurance to verify the design or the quality of the PORV if no guides or standards are available.

Although the TMI-2 PORV was not classified as safety-related and thus did not have all of the controls and standards applied to it as did other valves (such as the pressurizer code safety valves), it did have some good features. From information supplied by Dresser (as noted in Attachment 2), the TMI-2 PORV valve is a mature design with a large number of similar valves produced over a span of about 14 years; over 600 units have been produced for non-nuclear plants and about 30 units have been produced for nuclear units. Those produced for nuclear plants had harder materials used on valve disc and valve seat, and some quality assurance program controls applied. That the design is mature is best described by the Dresser spokesman who cited that only two design improvements have had to be incorporated.

SCOPE OF THE INVESTIGATION

The malfunction of the PORV is the endpoint of a long chain of safety-related decisions that were not recognized as being safety-related. This investigation started with the design concepts, and went through all the steps leading to the eventual performance of the PORV -- design reviews, licensing, purchasing, performance testing, quality assurance, and reliability data -- and ending with in-service surveillance, operational procedures, and maintenance practices at TMI. In other words, this investigation attempted to follow the four classic steps of management: (1) design and planning; (2) construction and fabrication; (3) operation and performance, and (4) audit and verification.

The sections in this report are, of course, not as neatly divided because they are in support of the significant findings which are not evenly distributed among the four steps.

Material was gathered in visits and field trips to:

- TMI, July 6-7, with Dwight Reilly's Quality Assurance Team*
- King of Prussia, Pa., July 9, NRC Region I Office*
- Bethesda, Md., July 10, headquarters of NRC, Office of Inspection and Enforcement, East West Building*
- Bethesda, Md., July 11, NRC Office of Nuclear Reactor Regulation (NRR), Phillips Building*
- Bethesda, Md., July 12, MPA and Division of Systems Safety offices*
- Bethesda, Md., July 13, NRR/Quality Assurance, Phillips Building*
- Pittsburgh, Pa., July 23, Westinghouse, Nuclear Center
- Phoenix, Ariz., Aug. 6-7, Institute of Electrical and Electronic Engineers (IEEE)/SC-5 Nuclear Reliability Committee
- Lynchburg, Va., Aug. 14-15, B&W headquarters, Valve Performance, and Precursor
- Alexandria, La., Sept. 19, Dresser Industries, supplier of PORV.

In addition, a great number of telephone contacts and data-gathering was performed in the course of the preparation and drafting of this report. Throughout, in all our contacts and meetings, we experienced the utmost in cooperation and support of the efforts of this Commission.

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* Visits marked with an asterisk contributed most toward the review of the quality assurance program and minimally toward the PORV investigation.

The investigation led also to discussions with valve specialists, standards committees, reliability data centers, and review of hearing records and depositions of this Commission. The original puchase order, specifications, test requirements, and assembly drawings for the PORV were obtained from B&W.

ANALYSIS

GENERAL DESIGN CONSIDERATIONS

Light water reactors (LWRs) may experience pressure transients during operation as a result of a mismatch between the reactor power level and the electrical load demand. When the electrical load demand is less than the power generated by the reactor, an increase in pressure of the reactor coolant generally results. The reactor coolant system (RCS) is enclosed in the reactor coolant pressure boundary (RCPB). Breaching of the RCPB due to overpressure (resulting in ruptures, large or small) must be prevented under all circumstances. All LWRs are designed with a pressure relief system, limiting RCS pressure under normal and abnormal pressure transients to 110 percent of design pressure. These design functions are accomplished through the use of a plant specific combination of safety valves, usually referred to as ASME code valves, and pilot-operated relief valves. In some boiling water reactors (BWR) plants, dual function safety/relief valves (SRV) are employed. A typical safety valve is shown in Figure 6, and a typical PORV (Dresser design) is shown in Figure 1. Safety considerations do not permit the safety valve to be isolated by a block valve. This restriction does not apply to the PORVs, because they are not considered safety valves and do not receive credit for their pressure-relieving capacity in required safety analysis reports. <u>The fact that PORVs are not safety valves does not mean that they are not related to safety.</u> An excellent source of design information is an Oak Ridge design manual, "The Selection and Procurement of Pressure Relief Valves for LWR Systems" (reference 5).

It is conventional to locate at least two safety valves (for redundancy) on the pressurizer, and one PORV to provide for convenience of operations by relieving pressure so that the code safety valves are not called on to operate. When code safety valves operate, the plants generally have to shut down to allow rework of the valves.

The need for PORVs is clearly associated with the expected frequency of transients and their severity. BWRs require several PORVs (reference 4). PWRs of certain design may do without PORVs. For example, the Palisades reactor of Consumers Power has operated since 1972 with a permanently blocked PORV. All existing Consumer Engineering reactors have their PORV and reactor trip activated by the same signal at the same setpoint. Westinghouse uses PORVs of a different design (see Figures 4 and 5). Based on available performance records, their PORVs are more reliable in operation and are easier to monitor as to open/closed position. B&W, on the other hand, has selected the PORV type shown in Figure 1 because it will not allow air-activated devices like the Westinghouse PORV in the nuclear steam supply system (NSSS). The B&W control system for the activation of the PORV is shown in Figure 2.

The PORV at TMI-2 was set to open at 2,255 psig, and to close at 2,230 psig. These setpoints are dialed into the non-nuclear instrumentation (NNI) cabinet, located in the cablespread room under the control room. The control systems, in general, are the responsibility of the architect/engineer, Burns and Roe. However, the NNI cabinet was manufactured by Bailey Meter Company under contract to B&W.

How does the PORV work? A pressure sensor in the hot leg of the pressurizer sends a signal to the safety-grade reactor protection system. The electric signal, representing the pressure, feeds into the NNI console which is essentially a computer following instruction punched into the system. The switch in the control room is also wired into the NNI (three positions: manual, off, and automatic). The NNI output goes to a bistable (signal monitor), which activates the relay (switch), which causes the PORV to open or to close. The discharge of the safety valves and the discharge of the PORV are piped together into the reactor coolant drain tank, provided for flow from relief valve operation. However, a substantial quantity of coolant will rupture the drain tank's rupture disc, as it did at TMI-2 and at the Davis-Besse event of September 1977. During normal PORV operation, RCS steam pressure enters the main valve through chamber A (see Figure 1) and passes upward around the disc guide in chamber B. Steam also enters chamber C through a clearance space between the main valve disc and the disc guide. The main valve disc is held in the closed position by the steam pressure in chamber C. PORV actuation is accomplished by energizing the solenoid in the pilot valve assembly. When the solenoid is energized, the pilot valve opens and allows the steam in chamber C to be vented to atmosphere through port F. The resulting differential pressure on the main valve disc causes it to open, thereby permitting steam to escape from chamber B to the valve outlet. The electrical control system for a PORV is designed to provide power to energize the pilot valve solenoid in any one of the following ways:

1. When the reactor system pressure reaches the valve setpoint, a pressure-sensing switch senses the pressure, actuates, and provides power to the pilot valve solenoid.

2. The reactor operator can open manually the valve, either for testing purposes or to relieve system pressure during a transient, by providing power to the pilot valve solenoid through a switch on one of the control room panels.

When power is removed from the pilot valve solenoid, the valve is supposed to return to the closed position.

<u>PORV DESIGN REQUIREMENTS</u>

The complexity of today's nuclear steam supply systems (NSSS) requires careful and systematic designs, design reviews, and system specifications, including definition of interfaces between responsible parties (B&W and Burns and Roe), as well as between overlapping functions, such as mechanical and electrical requirements. System requirements specifications (SRS) describe in detail the B&W NSSS (reference 6). The PORV specifications are contained in SRS-reactor coolant system.

The purpose of the reactor coolant system SRS is to delineate and to tie together the information necessary to (1) provide necessary design requirements for the procurement of a particular reactor coolant system component, such as a PORV; (2) provide requirements for (a) the interfaces between various components within the RCS, (b) the interfaces between the RCS and other systems within the B&W scope of supply, and (c) balance of plant (BOP) interfaces; and (3) assure that the design of the RCS components accurately reflects the analysis done to support these component designs or the design of the NSSS as a whole.

In addition to these functional requirements are regulatory requirements derived from 10 CFR 50. The following sections apply to PORVs:

- Section 50.2 -- Definitions (Definition V)
- Section 50.55a -- Codes and Standards
- Appendix A -- General Design Criteria (GDC) for Nuclear Power Plants (for a component-by-component listing of applicable GDC, see Table 2)
- Appendix B -- Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
- Appendix G -- Fracture Toughness Requirements
- Appendix H -- Reactor Vessel Material Surveillance Program Requirements

The following GDC are applicable to the PORV: 1, 2, 3, 4, 13, 14, 30, 31, 32, and 52. GDC-l is the requirement for applicable quality assurance defined in more detail in Appendix B's 18 criteria.

It is clear that safety-related structures, systems, and components deserve more attention in design, manufacture, construction, and operation than would be necessary for non-safety-related items. More attention means higher cost, often considerably higher, because of extra design and review efforts, testing, and qualifying; in other words, the application of an appropriate quality assurance effort. The difficulties arise in establishing a list of safety-related items (Q-list) and the "appropriate" quality assurance effort.

There is a subtle <u>inconsistency between Appendix A and Appendix B</u> (Quality Assurance Criteria for Nuclear Power Plants). The difference is in the wording. Appendix A speaks of "structures, systems, and components <u>important to safety;</u> that is structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public" (emphasis added). Appendix B is quite specific in its requirements of what has to be done in connection with items that "prevent or mitigate the consequences of postulated accidents." The best example is the primary coolant pressure boundary that must meet the ASME code requirements of Section III. The PORV is not considered to be a part of this boundary.

TABLE 2: **General Design Criteria Matrix**

This matrix identifies which General Design Criteria from Appendix A of Title 10 of the Code of Federal Regulations shall be considered applicable in the design of the various reactor coolant system components.

Criterion Number	1	2	3	4	13	14	15	19	26	27	28	29	30	31	32	34	44	52
Reactor Vessel & Closure Head	X	X		X		X							X	X	X			X
CRDM Service Structure	X	X																
Pressurizer Vessel	X	X		X	X	X							X	X	X			X
Pressurizer Heaters	X	X	X	X	X	X	X	X					X	X	X	X		X
Reactor Coolant System Pipes	X	X		X		X							X	X	X			X
OTSG	X	X		X	X	X							X	X	X			X
Supports & **Restraints**	X	X		X														
Core Support Assembly	X	X		X														
CRDM	X	X	X	X	X	X	X		X	X	X	X	X	X	X			X
RC Pump	X	X		X	X	X							X	X	X		X	X
RC Pump Motor	X	X	X	X													X	X
Pressurizer Spray Valve	X	X	X	X	X	X							X	X	X			X
Pressurizer Spray Iso-**lation Valve**	X	X	X	X	X	X							X	X	X			X
Pressurizer E/M Relief **Valve**	X	X	X	X	X	X							X	X	X			X
Pressurizer E/M Relief Isolation Valve	X	X	X	X	X	X							X	X	X			X
Pressurizer Safety Valves	X	X		X		X							X	X	X			X

In practice, some structures, systems, and components are more important to safety than others and, therefore, it seems appropriate that quality assurance measures should be graded in accordance with the safety functions. How to do this in practice is a tale of confusion and frustration that has yet no ending. It involves several technical societies (ASME, IEEE, American National Standards Institute, American Nuclear Society), utility organizations (Edison Electric Institute, American Public Power Association, EPRI), and, of course, the NRC. It also involves the availability of a data bank that can provide the designer and the operator with reliability data (for structures, systems, and components of interest in this dispute). Hence, it involves the licensee event report (LER) list and the other data bases.

A good summary of where this Appendix A/B issue now stands is contained in a recent NRC memorandum (reference 7) discussing "Applic-ability of the quality assurance criteria of Appendix B to structures systems, and components of nuclear power plants." A technical staff analysis report, "Quality Assurance," describes the issue (reference 8).

CODES AND STANDARDS

Section 50.55a stipulates certain applicable codes and standards among which the ASME code, Sections III and XI are best known. Section 50.55a-(f), "Valves," and (g), "Inservice Inspection," requirements are applicable to the PORV specifically. As a further help to applicants for construction and for operating license, the NRC issues regulatory guides that do not have the force of regulations but are a vehicle for promulgating acceptable practice and valid standards.

A comparison of the provisions of the ASME Boiler and Pressure Code, Section III, with the NRC requirements has been made; this relates the requirements to the N-stamped PORVs observed in the shipping area of Dresser Industries on Sept. 19, 1979, (see Attachment 3) to the NRC requirements.

In practice, the regulatory guides will make use of available standards, noting exceptions and additions. But when standards are not available, the NRC is obliged to generate a regulatory guide. This is viewed by industry as taking the place of a standard, although it is advisory only -- it is written with "shoulds," not with "shalls."

The nuclear standards effort in this country is organized under the Nuclear Standards Management Board (NSMB) of the American National Standards Institute (ANSI). NSMB consists of representatives from producers, consumers, and regulators; i.e., from utilities, architect engineer firms, reactor vendors, technical societies, NRC, DOE, etc. NSMB does not write standards per se, but stimulates and coordinates the various standards-writing organizations. NRC's Office of Regulatory Standards is represented on NSMB. Many large organizations, such as the ASME, IEEE, American Society for Testing Materials (ASTM), and ANS, develop standards in their field of competence and follow the ANSI approval process before these standards are promulgated (see Appendix A).

The best way to put lessons learned into practice is via the standards route. The development of boiler standards in the early 1900s, after many fatal boiler explosions, is a classic example of this lesson-learned dictum. Over the past 70 years, the boiler standards developed into the present ASME code.

The NSMB is aware of the importance of standards and their practical usefulness in all phases of nuclear plant design, construction, and operation. Its planning committee hopes to aid the recovery from the TMI accident by reviewing and updating of applicable standards. In its July 9, 1979, meeting, the planning committee commented on valve standards in general and on PORV standards in particular:

> The valve industry is fragmented; furthermore, the requirements for application of nuclear valves may not lead to use of valves meeting high standards in some areas in the plant where they are in fact needed (reference 9).
>
> The following standards need review and approvals:
>
> - N278.1-1975 (Functional Specification Standard for Self-Operated and Power-Operated Safety Related Nuclear Valve Systems.)
> - N278.2 (Proposed) (Functional Qualification of Self-Operated and Power-Operated Safety Related Nuclear Valve Systems.)
> - OM-76-7 (Proposed) (Performance Testing of Safety Valves.)
> - N278.3 (Proposed) (On-Site Testing of Self-Operated or Power-Operated Safety Related Nuclear Valve Systems.)
> - N278.4 (Proposed) (Nuclear Valves.)
> - E1-14T-1972 (RDT Standard. Pilot-Activated Safety Relief Valve. RDT standards apply to the old AEC and now to DOE construction. This standard is outdated and not used. No updating or reissuing is planned at present.)

In addition, an application standard may be required for PORVs; i.e., application of valves in systems whose interaction with safety systems may cause transients that challenge the safety systems.

In considerations of upgrading the PORV classification to "safety grade" and the associated controls and instruments to new standards for control systems, the NSMB suggests (reference 9) modification of three standards: N278.1-1975, P/N278.3, and B16.41 (Appendix B).

It is difficult to reach agreement between the various standards-writing organizations making up NSMB with regard to a consistent set of acceptable valve standards:

> Note that there is to date no evidence that safety-related relief valves are any more reliable in reseating than are

nonsafety grade valves. For example, there was difficulty for a long period with spurious actuation and failure to reseat of BWR safety relief valves to meet the ASME code. It is also well known_ that fossil plant safety valves commonly leak after they have lifted and must be reworked to make them tight.

It is not clear that any standard action is required to change the operational procedure so that the PORV is isolated during normal operation (reference 9).

A good indication of disagreement between ASME and IEEE approaches is reflected in Attachment 1.

INSPECTION REQUIREMENTS

B&W placed a purchase order (#022660LS) on Dresser Industries, dated Oct. 1, 1970, for two pressurizer safety valves and one PORV. The specification for the PORV (CS-3-79/NSS-6/0170, dated Jan. 1, 1970), was later incorporated into B&W's system requirements specification (SRS). The same applies to their quality control program specification (1132/0369, dated March 19, 1969). Over the intervening years, many new inspection requirements were added, but are not applicable to this PORV at TMI-2.

An inspection report by B&W, dated March 13-14, 1973, indicates satisfactory results of inspection. Type of inspection: final assembly functional test, and wall thickness minimum dimensions (see Attachment 4).

Dresser was also audited by Met Ed under the CASE system, Aug. 4-5, 1977 (Appendix C). No deficiencies were found. Dresser produces nuclear valve bodies under ASME code requirements. However, an N-stamp does not guarantee functional performance. The difference between quality control (material) and qualification (function) has engaged the industry for some time. The IEEE has accepted lead responsibility for qualification standards, including electromechanical components like the PORV. No such standard has yet been forthcoming (Appendix D). The ASME has accepted the least for all quality assurance standards.

Inspection requirements during installation of the PORV are practically nil, since the PORV was not considered to be safety-related. The same applies for inspection requirements during operations. But here, the history of valve failures sparked some progress in the ASME code.

Section XI, Inservice Testing, contains "Scope and Responsibility." The IWV-1100 Scope (as revised in the winter 1977 addenda) reads:

> This subsection provides the rules and requirements for inservice testing to certify operational readiness of certain Class 1, 2, and 3 valves (and their actuating and position indicating systems) in light water cooled nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of an accident.

However,. many safety and relief valves in class 2 and 3 systems are not specifically used in shutdown or mitigating the consequences of an accident, but if they failed under overpressure conditions, a system failure could result. Hence it was suggested to add to the scope six additional words: ". . or which provide overpressure protections" (see Attachment 5).

Standards-writing groups that might have been engaged in the development of an appropriate standard are literally turned off from their task by the premature issuance of a guide in the area of their expertise. Their task remains incomplete until a problem in the application of the regulatory guide forces the issue to be reconsidered at the level where quality standards are produced: namely, in the competent, experienced, diversified, and dedicated writing group.

Apparently, the NRC saw the need to draft a regulatory guide on "Recommendations for Inservice Testing of ... Safety System Valves," including a value/impact statement (Appendix E). The draft points out that:

> the Code does not address the issue of placing a plant in an unsafe condition as a result of valve testing during plant operation; i.e. if a valve fails in a non-safety position during exercising in accordance with the Code requirements, and causes a loss of systems functions. There is also a need to develop a full understanding of the impact of a decreased test frequency on valve availability and on system reliability, and it is not apparent that the possibility of damaging valves by testing was considered in establishing the requirements of the Code. A cursory review of LERs shows that the incidence of valves not meeting functional requirements increases with increasing life of the system.

Other applicable regulatory guides, as of March 1979, are tabulated below:

- Pressurizer Safety Valve -- 1.7, 1.26, 1.28, 1.29, 1.31, 1.37, 1.38, 1.44, 1.48, 1.61, 1.66, 1.84, 1.85, 1.92, 1.116, and 1.123

- Remote Operated Valves (PORV) -- 1.7, 1.26, 1.28, 1.29, 1.30, 1.31, 1.37, 1.38, 1.44, 1.48, 1.54, 1.58, 1.61, 1.64, 1.66, 1.73, 1.74, 1.84, 1.85, 1.89, 1.97, 1.106, 1.116, and 1.123

PORV FAILURES OTHER THAN IN B&W UNITS

On Jan. 11, 1973 -- please note the date -- Nucleonics Week (Vol. 14, No. 2) carried this story:

> The [Atomic Energy Commission] AEC is taking immediate steps to tighten up preventive procedures relating to inadvertent primary coolant release following a study of eight cases of

such releases from BWRs. In each case the release was caused by malfunctioning safety and relief valves that either opened prematurely or opened properly but then failed to close. The AEC concluded that attention should be given to determining the basic causes of safety valve malfunctions, and to the timely development of plans to prevent or minimize occurrences of coolant release.

Five years later, in July 1978, the NRC issued a technical report on "Operating Experience with Boiling Water Reactor (BWR) Pressure Relief Systems" (reference 11).*

Since the mid-1970s, manufacturers of valves and the General Electric Company have been working with the NRC and ASME to improve pressure relief valves. The need for better reliability and hence better availability of plant became ever more significant as the size (and cost) of plants increased. Replacement power cost for a TMI-2-sized plant is over $300,000 per dam. Now, an increasing number of BWR valves are periodically inspected, tested, and refurbished. Consequently, reported failure rates decreased dramatically:

1974: 1 failure per 40,000 hours of operation.

1978: 1 failure per 80,000 hours of operation.

In comparison to BWRs, PWRs (pressurized water reactors) in general are more sluggish in their response to feedwater transients. Now, of the three PWR nuclear steam supply system vendors, only B&W uses the OTSG (once-through steam generator), allowing for a more rapid response to transients, as compared to Westinghouse or Combustion-Engineering systems. Hence, among the three PWR vendors, B&W ought to be most interested in the reliability of safety and relief valve performance.

The statistics of feedwater transients in PWR systems reflect this situation. The NRC staff reviewed feedwater transients in PWR plants during the period from March 1978 to March 1979 (reference 4). There were nine B&W plants that had 27 feedwater transients, or 3.0 per year per plant. Twenty-four Westinghouse plants had 44 transients (1.8 per year per plant), and 7 Combustion-Engineering plants had 13 transients (1.8 per year per plant). Not every transient activates the PORV. But, clearly, there are more activations probable in B&W nuclear plants.

Several non-B&W system PORV incidents are reported. The most important events involving a stuck-open PORV are:

- Westinghouse -- Beznau, Switzerland, Aug. 20, 1974. Cause: fractured yoke (see Attachment 6).
- Combustion-Engineering -- Palisades, Sept. 8, 1971. Cause: loss of power (see Attachment 7).

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* NUREG--462 summarizes BWR valve failures between 1970-1978.

TABLE 3: Boiling Water Reactor Valve Failures

Type of Event	Safety Valve	PORV	Total
Inadvertent Blowdown (failed open)	49	4	53
Failure to Open	11	16	27
Potential Failure to Open	16	1	17

A comparison of primary thermal and hydraulic parameters between B&W, Westinghouse, and Combustion-Engineering nuclear steam supply systems is given in Table 4. In early April 1979, a task group chaired by Ashok Thadani of the NRC reviewed and reported on Westinghouse and Combustion-Engineering operating plants in the light of the TMI accident. The report contained an assessment of the generic aspect of feedwater transients and the related ensuing events at TMI-2:

> Events at TMI-2 have shown that under certain circumstances heavy reliance may need to be placed on nonsafety-grade equipment to shut down a plant. The staff should review the impact of the use of nonsafety-grade equipment for shutdown, including control systems and PORVs. (reference 12)

Data from operating U.S. Westinghouse plants shows that the PORVs have opened approximately 60 times during normal operation for various reasons. For each of these openings, the valve reseated correctly. In most cases the Westinghouse data does not include the pre-operational testing phase predominant in reported B&W failures.

A failure of a PORV to reseat fully in a Westinghouse unit was recently reported at McGuire-1, which was in hot functional testing. The malfunction was the result of the valve plug binding in the valve bonnet recess area. The exclusion of this type of data from the data base affects the calculated frequency of stuck-open PORVs for Westinghouse units (reference 12).

PORV FAILURES IN B&W UNITS OTHER THAN TMI-2

Reviewing B&W records of PORV failures to operate correctly in B&W plants has resulted in the identification of nine such events prior to the TMI-2 accident. Seven of nine failures are failures in the open position (Table 1). (Additional information is contained in Appendix F.) All B&W reactors use Dresser valves except Davis-Besse-1, which employs

TABLE 4: Comparison Of Primary Thermal-Hydraulic Parameters

Vendor		B&W				Westinghouse		C-E
Reactor	TMI-2	TMI-1	Rancho Seco	Davis-Besse	Oconee 1	H.B. Robinson	Trojan	Calvert Cliffs 1 & 2
Design power, MWt	2772	2568	2772	2772	2568	2190	3411	2560
T_{in} °F	557	554	557	555.4	554	546.2	552.5	543.4
Tout Core, °F	610.6	606.2	610.6	611.7	606.2	604.5	619.4	597.4
Tout Vessel, °F	607.7	603.8	607.7	608.6	603.9	602.1	616.7	595.4
Core pressure, psia	2200	2200	2200	2200	2200	2250	2250	2250
Core flow 10o lb/hr	137.8	131.32	137.8	131.32	131.32	101.5	126.7	117.5
Core flow area, ft2	49.17	49.17	49.17	49.17	49.17	43.75	51.1	53.5
HPI Injection pressure, psia	1615	1515	1615	1615	1500	1715	1765	1578
Cooland subcool-ing at injection pressure, of	24.0	18.8	24.0	24.3	17.4	40.4	34	33.8
Subcooling at core outlet normal, of	39.0	43.4	39.0	37.9	43.4	48.4	33.5	55.5

r
N

a Crosby PORV. This B&W tabulation (Table 1) is at variance with NUREG-0560 (reference 4), which reports five occurrences as shown in Table 3.

One of these five, the September 1974 event at Arkansas Nuclear One, cannot be located in any B&W file. It is more fully described in Attachment 8. Of the nine events listed in Table 2, two are of major interest because the events occurred during operation, and the PORV stuck open. These two events, Oconee-3 of June 1975 (12 percent power) and Davis-Besse of September 1977 (9 percent power), are described in more detail in this section.

Another event not included in the B&W files of stuck-open PORVs is the Rancho Seco event of June 1978, which was included in the summary of NUREG-0560 as one of five events with an open PORV; however, this "open" event was classified as leakage.

Of the 11 stuck-open PORV events, prior to TMI-2 of March 28, 1979, nine were in B&W plants. Of these nine stuck-open events, B&W records included seven.

Oconee-3 Incident, June 13, 1975

In the course of a routine maintenance shut down, a minor system transient occurred, which resulted in opening the PORV on the pressurizer. The reactor power had been reduced to approximately 15 percent, unit load demand was 65 MWe, and power generation was 115 MWe. This difference existed because the reactor was operating automatically at its low limit and could not further follow unit load demand. Operator placed the turbine control station on manual, leaving the integrated control system (ICS) in the load-tracking mode. This resulted in oscillating RCS conditions and hence led to a temperature and pressure transient.

The PORV had opened at the set point of 2,255 psig, but failed to close when pressure dropped below 2,205 psig. The open/closed lights in the control room did not indicate that the PORV was still open. Pressure dropped further, the reactor tripped on low pressure, and the HPI system actuated.

The reactor operator closed the PORV-isolation valve (block valve) immediately after reactor trip, but soon reopened it because of the rapidly rising pressurizer level. The block valve was finally closed again when the RCS pressure reached 800 psig, terminating the pressure transient.

The subsequent controlled cooldown of the RCS, when combined with the temperature drop during the transient, resulted in a cooldown in excess of the allowed technical specifications. The transient and associated events also caused the quench tank rupture disc to blow open, and the release of about 1,500 gallons of reactor coolant to the reactor building sump.

It was found that the PORV was stuck in the open position because of heat expansion, boric acid crystal buildup on the valve lever, rubbing of the lever against the solenoid brackets, and bending of the solenoid

spring bracket. The malfunction of the valve position indication was apparently caused by the sticking of the solenoid plunger at slightly less than the full open position, or by crud buildup around the switch for the open/closed indicator lights.

To prevent recurrence, the following corrective actions were implemented at Oconee, but not at other plants:

- **The unit shutdown procedures were revised to include a change that would prevent decreasing unit load demand below 120 MWe before placing the instrument and control system (ICS) in the tracking mode.**

- **The TMI-1 and -2 PORV valves were examined for any indication of boric acid crystal buildup and reconditioned.**

- A **test to cycle RC-66 prior to startup with a test signal corresponding to 2,285 psig, was to be incorporated into the station operating procedures.**

TABLE 5: B&W PORV

Reactor	Date	Assigned Cause	Reactor Condition
		Electrical Failures	
Oconee-2	8/73	Wiring error	pre-operational
Crystal River	11/75	Stuck solenoid	pre-operational
Davis-Besse	9/77	Missing relay	9% power
TMI-2	3/78	Loss of power	zero power physics
TMI-2	9/78	Failed to open	unknown
		Mechanical Failures	
Oconee-2	11/73	Pilot leakage	start up
Oconee-3	6/74	Vent failed to open	**pre-operational**
Oconee-3	6/75	Corrosive leakage	12% power
Davis-Besse	10/77	Pilot clearance	hot standby
		Unknown Cause	
TMI-2	3/79	? ? ?	97% power

TABLE 6: Power-Operated Relief Valves on Pressurizer for BMW Plants

Valves for B&W Plants	Arkansas 1	Crystal River 3	Davis-Besse-1	Oconee 1	Oconee 2	Oconee 3	Rancho Seco	TMI-1	TI-2
Power-Operated Relief Valves									
Mfg.	Dresser	Dresser	Crosby	Dresser	Same	Dresser	Dresser	Dresser	Same
Number	1	1	1	1		1	1	1	
Type	Electromatic	Electromatic	Electromatic	Electromatic		Electromatic		Electromatic	
Model no.	31533VX-30	Same	HPV-ST	31533 VX-30		31533VX-30		31533VK-30	
Size	231" x 4"		2'" x4"	2331" x4"		211" x4"		2'" x4"	
Relief cap.	106,450 #/hr	100,000 #/hr	112,000 #/hr	100,000 #/hr		110,000 #/hr		106,450 #/hr	
Set press.	2,300 psig		2,235 psig	2,300 psig		2,300 psig		2,300 psig	
Reseat press.						2,220 psig		2250 psig	
Malf. date (Significant) cause)	9/1/74 Improper venting	None	9/24/77 Steam pilot valve system	None	None	June 1975 Boric acid crystal buildup bent lever on pilot valve	June 1978 Valve leakage	None	3/29/78 De-ener vital b
Fail position	Closed (Class 1E)	Closed (1E)	Closed (non-1E)	Closed (non-1E)			Closed (non-1E)		
Position indicator	Yes (Pilot-red/green)	Yes (open-closed)	Yes (on pilot-red/green Lights)	Yes (open-closed)			No	Pilot-red/green	
Thermocouple indicator and alarm	Yes (computer)	Yes (computer)	Yes (computer)	Yes (computer)			Yes (computer)	Yes	
Thermocouple type and location	Strap-on	Well/-90 ft from valve	Strap-on/ -1 ft	Strap-on/6 7 ft downstream			Strap-on/40 ft from valve		
Block Valve									
Mfg.	Velan	Dresser	Velan	Westinghouse	Same	Same	Velan		
Type	Motor-operated	Motor-operated	Motor-operated	Motor-operated			Motor-operated	Moter-operated	Same
Fail position	As-is (Non-1E)	As-is (Non-1E)	As-is (Non-1E)	As-is (Non-1E)			As-is (Non-1E)		
Position indication	**Yes**	Yes	Yes	Yes			Yes	Yes	

Source: NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company," May 1979, Table 7.

- Operating personnel were advised of this incident with specific instructions that immediate closure of the block valve is the proper corrective action for such occurrences.

In accordance with standard technical specifications, an abnormal occurrence report was submitted on June 25, 1975. This event is listed under LER 750IT on the LER list (reference 13) of valve events. None of the other events listed in Tables 5 and 6 could be found on this list.

Davis-Besse-1 Incident, Sept. 24, 1977

This event was similar in many respects to the TMI accident, except that the reactor at Davis-Besse was operating only at 9 percent power versus 98 percent at TMI. A feedwater transient (through several intermediate steps) resulted in the PORV lifting. The valve then rapidly oscillated closed/open approximately nine times and remained in the full-open position. A temperature rise in the primary system caused an increase in the pressurizer level, and the operator manually tripped the reactor because of high pressurizer level. The stuck-open PORV rapidly reduced RCS pressure, and the safety features actuation system (SFAS) tripped at 1,600 psig setpoint initiating the HPI system. The PORV discharge overloaded and over-pressurized the quench (reactor coolant drain) tank. About 4-1/2 minutes after reactor trip the quench tank rupture disc blew, thereby venting its content into the containment. Soon after this, the operators throttled the HPI pumps on the basis of increasing pressurized level even though pressure in the RCS was decreasing. Approximately 20 minutes after reactor trip, the operator diagnosed the reasons for the primary system depressurization, and closed the block valve to the PORV, terminating the blowdown of the primary coolant to the containment.

Subsequent operator action using make-up pumps and high pressure injection pumps stabilized the primary system pressure and pressurizer level and a controlled shutdown to cold conditions followed.

Failure of the PORV to close following actuation can be attributed to a missing seal-in relay. This relay holds the PORV open until reset pressure of 2,205 psig is reached. Because of this missing relay, the PORV oscillated (nine times) and eventually stuck open.

The Davis-Besse incident of September 1977 illustrates some faults in nuclear safety systems:

- There is no clear understanding of what are safety-related components or actions.

- There is no mechanism for prompt and effective corrective actions.

- There is no forum for failure mode and effects analysis (FMEA), for failure incidents reviews (FIR), or trend analysis.

Rancho Seco Accident, March 20, 1978

Another accident of considerable similarity and significance occurred about the same time as the TMI incident. This event involved the non-opening of the PORV upon power failure in the NNI. At the time of the event, the PORV was isolated by its block valve because of prior execessive leakage. Current Events/Power Reactors (reference 14) describes the accident:

> A light bulb was inadvertently dropped into the open light assembly cavity during replacement of a burned-out bulb. This created a short in the "Y" portion of the 24-volt DC NNI buses. During the resulting current surge the protection for the DC power supply actuated, causing approximately two-thirds of the NNI signals (pressure, temperature, level, flow, etc.) to provide faulty information to the control room and to the ICS.

The instrument and control system (ICS), attempting to match equipment output to the erroneous signals, reduced feedwater flow to zero. This reduction caused the RCS pressure to increase, as happened in the TMI-2 accident. Instead, the reactor tripped at its higher trip setting. When the power supply was restored one hour and 10 minutes into the transient, the RCS temperature had dropped to about 285°F, beyond the technical specs limits, and pressure to 1,400 psig. During this transient, it was difficult for the operators to ascertain which of their indications were valid. The plant had to be controlled using the select few parameters that were known. Another effect was that the equipment was operated automatically, without regard to actual conditions, since spurious signals were fed into the ICS. An NRC memo (reference 15) concludes:

> Although the actual safety implications of this particular transient were minimal, this is only true because it occurred very early in the plant life. We strongly recommend that positive steps be taken to prevent transients of this kind, and that generic implications be reviewed promptly.

PORV FAILURES AT TMI-2

On March 29, 1978, one day into their zero-power physics program, an incident caused the PORV to open and remain open. The cause: loss of power to the solenoid actuator.

While performing reactor coolant pressure boundary (RCPB) isolation and cooling surveillance testing, the vital bus which feeds the RCP1A monitoring circuit was de-energized (blew a fuse). Since RCP2A was already down for clutch **repair, the loss of power de-energized the non-nuclear instrumentation (NNI). Because of loss of the NNI bus, the** PORV received an open command which initiated a rapid system depressurization. For corrective action the architect engineer (Burns and Roe) made field changes in the circuitry so as to have the PORV remain closed upon loss of power to the NNI bistable (see Attachments 10, 11).

The question remains unclear of the original design intent as to whether the PORV controls at TMI-2 were intended to fail open or closed. The fairly complex interaction of electrical control power and NNI electrical power to the bistable is a case where the application of an FMEA (failure modes and effects analysis) would help the designers identify the problem failure modes. There were several followup actions on this "failsafe issue."

- Followup by NRC. A memorandum from I&E (Inspection and Enforcement) Inspector Sternberg to K. V. Seyfrit (Appendix G) requested "that the adequacy of the design approach (i.e., valve failing open on loss of control power) be reviewed on an expedited basis for B&W facilities in general and TMI in particular." Seyfrit responded on May 3, 1979: "The request is based on failure of the valve in the open position. Failure in this position is covered in Section 7.4.1.1.6 of the FSAR [Final Safety Analysis Report]. We conclude that additional review is not warranted." However, the reference to the FSAR is not appropriate. The FSAR does not address the "adequacy of the design approach" to failsafe conditions. It reads: "In the event that the relief valve were to fail in the open position, pressure relief could be controlled by cycling (open and close) the relief isolation valve" (reference 13). NRC does not require a clear statement in the SAR (Safety Analysis Review) of the designer's intent to have the valve fail open, or closed, upon failure of electric power.

- Followup by B&W. The site report suggested that no further action was required: "At other B&W plants, the PORV remains closed upon loss of power." But, later, in the post-TMI period, B&W service managers contacted each of their operating customers concerning the failing open of the PORV on loss of power either to the valve proper or to the electrical circuitry which controls the valve (see Attachment 9).

  The following responses were recorded:

SMUD	-- fails shut (Rancho Seco)
Oconee	-- does not open
Davis-Besse	-- does not open
Arkansas Nuclear One	-- if closed, will not open; if open, may not close
TMI-1	-- if closed, remains closed; if open, remains open on loss of power to control circuitry
FPC	-- fails shut (Crystal River)

  The memorandum continues, "It is not apparent that even if the valve does stay open that it is completely bad. It provides a relief path thus preventing the opening of the safety valves and can be controlled by use of the block valve." There is no indication that B&W has adopted a "failsafe criterion" for PORVs. The Fairburn memor.Lldum also reflects the classic position that PORVs are not safety-related and are important for operational convenience.

Other PORV Startup Problems at TMI-2

B&W lists another PORV failure that occurred in September 1978 during the cold startup procedure (see Appendix F, Incident 9). In this case the PORV failed to open at the prescribed setpoint due to electric malfunction.

This event is reported by General Public Utilities (GPU) SPR (startup problem report) 2816 (see Appendix H). All together, five GPU SPRs are appended referring to PORV startup problems at the following dates: Aug. 18, 1977, GPU-5055; Aug. 29, 1977, GPU-5072; Aug. 29, 1977, GPU-5073; and Oct. 5, 1977, GPU-5147.

There is no assurance that all PORV startup problems are recorded and documented. Four out of the five GPU SPRs above are indexed to the RCS, and not specifically to the pressurizer or to the PORV.

CONFIGURATION CONTROL AT TMI-2

The March 1978 event involved the reactor coolant system which is a B&W responsibility, and the instrument control system, which is a Burns and Roe responsibility.

How are their functions coordinated? During the design-phase of TMI-2, B&W collects all balance of plant (BOP) requirements stemming from their nuclear steam supply system design and transmits this package to Burns and Roe via the utility customer. This function of collecting is now performed by the Plant Integration Group at B&W -- an organizational unit that came into being only about 1974. It was previously done in the project manager's office. Some B&W requirements are very significant (for safety or operational reasons), are usually double asterisked (**), and should not be changed by the architect engineer without prior approval by B&W. There is no indication that this rule can be audited (enforced), nor that the PORV requirement was so identified(**).

In fact, the PORV requirement was transmitted as a drawing (see Attachment 12) which is not too clear as to what is intended as the failsafe condition. A clearer drawing (see Attachment 13) of the control logic was later transmitted. Burns and Roe followed through with a letter to the GPU Service Corporation, dated Sept. 1, 1977.

Figure 7.6-7 in the Final Safety Analysis Report (FSAR) represents the PORV control logic as it was wired at the time of the March 1978 accident (see Attachment 14).

On April 6, 1978, Burns and Roe field engineers drafted an engineering change memorandum (ECM) to provide "open signal on" indication for the PORV and to revise logic to have the PORV "fail close" on loss of power (see Attachment 15). It also states that:

> B&W has reviewed the PORV logic and agrees to the concept of having the PORV fail closed on loss of NNI power supply to the hi-low Monitor. To achieve this condition switch S-1 should be in the de-energized mode and the wiring modification be made as indicated in the attached sketch.

The sketch (Attachment 15) corrects the FSAR drawing. It is not known whether the FSAR has been revised to the as-built and modified conditions. B&W's site problem report 183 is also included in Attachment 15.

This PORV failure is a clear indication of the need for better configuration control and interface coordination. It provides a good opportunity for a failure mode and effects analysis (FMEA). In the past, probabilistic risk assessment has rarely been applied to nuclear systems. The American Nuclear Society proposed in mid-1978 to develop criteria for the performance of probabilistic risk assessments of nuclear power plants and their systems. This effort was boosted further with the findings of the Lessons Learned Committee (NUREG-0578), which concluded that "explicit consideration should be given to the effects of a loss of onsite or off-site power" (reference 16).

The history of the PORV control -- as seen by Burns and Roe -- is contained in Appendix H (see also Attachment 11).

<u>ROLE OF THE PORV IN THE TMI-2 ACCIDENT</u>

The accident sequence is well described in NUREG-0600 (Inspection and Enforcement report) (reference 1). Significant events are that 3 to 6 seconds into the accident, the PORV opened when the RCS reached 2,255 psig setpoint. A few seconds later, the reactor tripped when the RCS reached 2,355 psig. About 20 seconds into the event, the RCS pressure decreased sufficiently to call for the closure of the PORV. At this time the light on the panel board so indicated. But unknown to the plant operators, the PORV did not close (for unknown reasons), passing reactor coolant from the top of the pressurizer, and became the reason for a LOCA.

It was not until after 2 hours into the accident that operators manually closed the block valve, stopping further loss of coolant. Until the valve can be examined, the cause of its failure to close will remain unknown. Even after examination, the cause may not be determinable because of the conditions that it has been subjected to since it failed.

To prevent a similar situation in other operating B&W reactors, and in support of NRC's Bulletin 79-05B, a letter was sent by B&W to all its customers (April 21, 1979) advising them to "reduce the pressure setpoint for a reactor trip at high pressure to 2,300 psig and additionally, increase the high pressure set point for the PORV to 2,450 psig." Both of these measures tend to preclude actuation of the PORV during anticipated transients, and still provide the operational convenience to protect the safety valves.

PERFORMANCE AND RELIABILITY OF PORV

In order to detect a trend, a generic issue, or an isolated event in the malfunction of the *PORV*, one must have access to a consistent, complete, and reliable pool of performance data. For the case of the *PORV*, this information (data bank, reliability figures, trend analysis) is not yet available.

Reliability data based for PORV performance are all limited as to their information input, and to the retrievability of needed information. The blame for this deficiency can be put on two perceptions regarding the PORV. First, it is not considered to be safety-related; and second, it is an active valve, an electro-mechanical device, that requires considerations of substance as well as of function -- quality testing as well as qualification testing (Appendix I).

Valves are, by far, the most failure-prone components in nuclear plants. The Rasmussen Report (WASH 1400) analyzed failure from 17 LWRs among 19 systems or components. In its Table III, the total failures recorded are 303, of which 102 are valve failures (1972 data). In other words, of all significant failures in LWR plants, one-third are due to valves. The contribution of the PORV to this record is unknown, since most *PORV* failures are not recorded. It is noteworthy that valves of all kinds are the most numerous components in any nuclear power plant. According to a current IEEE study, there are over 11,000 valves (1/2-inch or larger) in a present-day *PWR* plant.

An NRC document (reference 13), entitled "LER Output on Events Involving Safety/Relief Valves from 1969 to the Present," was reviewed for PORV failures. Output was sorted by facility and event data. Of the identified stuck-open PORV events, only one can be found in the above-cited LER output -- the Oconee-3 event of June 13, 1975.

LER outputs record four events per page. Hence, there are over 450 events involving safety/relief valves on the LER list. Since these compilations are not complete it does not seem worthwhile to categorize these events or use the list for other statistical purposes. The NRC's Lessons Learned Task Force 6 came up with some statistics: "Based on incomplete data, there have been five known instances, out of about 230 actuations in about 200 reactor years of service, of failure of a relief valve in a PWR to properly close" (reference 16). This report details 11 such events.

Part of the difficulty with the LER list is the ambiguity of reporting requirements. NUREG-0161 (reference 18) gives detailed instructions but cannot overcome the problem of "what is safety related?"

The General Accounting Office (GAO) investigated the LER system and other reporting functions and issued EMD-79-16, "Reporting Unscheduled Events at Commercial Nuclear Facilities: Opportunities To Improve Nuclear Regulatory Commission Oversight" (reference 19).

The GAO reviewed the NRC's program for collecting and evaluating licensees' reports of incidents or unplanned events. GAO found that the NRC needs to improve its licensee report assessment procedures to better assure that it is identifying and acting on all safety-related problems. For example, the NRC's review of reported events following its discovery of a safety-related problem at two operating nuclear power plants revealed that the problem had been widespread for some time. Better assessment procedures may have enabled the commission to identify this problem earlier. GAO also found that the commission should extend its licensee report requirements to types of events not now covered.

Nuclear Safety Information Center

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the Research and Development (R&D) Office of the NRC. The data base consists of evaluated LERs and appropriate technical articles in the open literature. "Operating Experience with Valves in LWR Nuclear Power Plants for the Period 1965-1978" (reference 20) reports on 1,842 safety-related events for BWRs and 1,685 safety-related events for PWRs. Of the latter, 135 are pressure relief valve events and 79 involve the pressurizer valves. Unfortunately, it is not possible to identify PORV events alone, and among these, the stuck-open events of interest in this study.

Complete and informative failure data on nuclear power plant equipment would be useful to NRC, and to those who design, construct, and operate these plants. They could result in:

- improved system design and better schedules for surveillance and tests;

- identification of failure trends and wear-out patterns;

- reduced time to license power plants; and

- improved maintenance and spare parts management, and component purchasing evaluation.

With such information, nuclear power plants could expect to have increased operational reliability which would result in increased on-line time. The GAO recommends to the NRC to use the rulemaking procedures to decide the issue of mandating full nuclear industry participation in the industry's voluntary reliability report system (reference 19). Many utilities have not made meaningful efforts to participate in the system. The NRC attributes this to uncertainty over the future of nuclear power. At this time, NRC is not convinced of the need to mandate full industry participation, because it does not believe any major nuclear power plant design improvements would result. The NRC intends to study the issue further while increasing its financial support to the voluntary system.

The following data bases and information sources were accessed:

- LER list at NRC (reference 13)
- Nuclear Safety Information Center (Oak Ridge) (reference 20)
- Nuclear Reliability Data System (Southwest Research)
- EPRI - National Electric Reliability Council (Princeton)
- IEEE/SC-5 Nuclear Reliability Committee, Project-500 (Appendix I)
- NRC Unresolved Safety Issues (NUREG-0510) (reference 21)
- B&W records
- Nuclear Standards Management Board, FIRR Committee of ANSI and FIRR (Failure and Incident Reports Review) (Appendix A)
- WASH-1400, the Rasmussen Report.

A detailed inquiry at the above information sources disclosed that today, there is no way to find accurate, pertinent, hard data on PORVs (or even on valves in general) to initiate a corrective action program. An integrated national nuclear reliability data bank becomes a national necessity. Both EPRI and the NRC are preparing for such data banks. It will be many years before either data bank will provide useful reliability data for PORVs.

The focal point for nuclear reliability is the Nuclear Reliability Committee of the IEEE. This committee met Aug. 6-7, 1979, in Phoenix, Ariz. The attached trip report (Appendix I) gives an insight into the status of the various projects concerned with nuclear reliability. The complexity of the subject of reliability engineering is illustrated in Table 7 from a proposed RDT Standard F2-9T on Reliability Engineering (Appendix J).

<u>Recognition of PORV Problems</u>

As noted in Table 1 and in Table 4, there is considerable history to the failures that have been experienced with the PORV by the various utilities that utilize them. Table 1 contains a list of 11 cases before TMI-2, where the PORV failed in the open position. Six of these failures had not been included in the recent NRC summary given in NUREG-0560 (reference 24). Eight of those failures were on valves supplied by one supplier, Dresser Industries; only four of these failures were known to the supplier (see Attachment 2). The 11 cases cited in this report may not be all, because, as has been noted, there has been no requirement for systematically reporting problems experienced with the PORV, principally because it has not been classified as safety-related equipment.

Because of this lack of systematic reporting, the accumulation of failure trend information on the PORV has not been accomplished; thus, its problems and ailments have not been highlighted. This has resulted in no concentrated failure analysis to determine the causes and appropriate corrective action to prevent recurrence of PORV failures.

TABLE 7: Guidance Literature to Reliability Tasks

TASKS \ REFERENCES	1 IEEE 352	2 RELIABILITY MANUAL FOR LMFBRS	3 NUC. PWR. REACTOR INST. SYS. HDBK VOL. 1	4 RELIABILITY HDBK IRESON	5 CODIER, ERNST O. 1968 SYM.	6 BJORO, EDWIN F. ASQC 1980	7 NOT F22 QUAL. ASSUR.	8 MIL HDBK 217B	9 MIL-STD-471A MAINTAINABILITY	10 MIL-STD-7MA PREDICTION	11 MIL-STD-781B REL. TESTS	12 MIL-STD-785A PROGRAM	13 IEEE STD 500 1977	14 WASH 1400 (NUREG - 75/104)
APPORTIONMENT	7.1			14.2								5.2.2		
FMEA	4.0	5.0 6.0	11-3									5.2.4	APP D	
MODEL	5.2	3.0	11-3					APP A				5.2.2		
PREDICTION	5.2	3.0	11-3	5				APP A		5.0		5.2.2		
TRADE-OFFS								APP A						
SENSITIVITY	5.6.2		11-3											
MAINTAINABILITY				11					ALL					
COMMON CAUSE FAILURES	4.5.2	7.0												APP IV
PHYSICS OF FAILURE		13.0		[illegible]										
TESTING FOR RELIABILITY		10.0 11.0	11-[illegible]	[illegible]							ALL	5.3		
DATA COLLECTION AND TREND ANALYSIS	6.3				PP 452 482							5.4.1	APP D	APP III
RELIABILITY CRITICAL ITEMS												5.2.5		
PARTS & MATERIAL CONTROL PROGRAM														
DESIGN REVIEW							3.5					5.2.7		
(PROGRAM PLAN)						PP 2-2 2-25						4.4		

NOTE: THE NUMBERS IN THE MATRIX REFER TO THE APPLICABLE PARAGRAPHS OF THE REFERENCED DOCUMENTS

One of the direct effects of such highlighting is to focus attention by the appropriate people in NRC and industry to bring about this corrective action; of course, this has not been done.

Examples of what has not been done to bring about attention to the PORV problem are given in Attachments 17, 18, and 19. The Advisory Committee on Reactor Safeguards has not recognized the PORV as safety-related or as a reason for initiating a generic item (see Attachment 17). NRC has not included malfunctioning PORV in the abnormal occurrences reported to Congress (see Attachment 18). NRC has not included PORVs in the unresolved safety issues reported to Congress (see Attachment 19).

SINGLE FAILURE CRITERION AND THE PORV

An article in Nucleonics Week dated June 14, 1979, stated that:

> Designing and building power plants so that they have a minimum probability of failing under improbable events does not guarantee maximum safety, and does not guarantee that the risk to the public is at the practically achievable minimum. It is more important to design and protect against the more likely events. (reference 22)

Others have found that the NRC staff has spent too much time in the past analyzing, testing, and researching low probability accidents. Much less time and money have been allocated to high probability events or transients.

The TMI accident demonstrates that the ritualistic application of the single failure criterion is not in the best interest of safety. Appendix A of 10 CFR 50 defines this criterion: "A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety function." It further explains that:

> Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive component function properly) and (2) a single failure of a passive component (assuming active component function properly), results in a loss of the capability to perform its safety function.

In the case of the PORV, the passive part is designed to meet ASME code pressure boundary standards. The active part -- the part that failed open -- was designed so that, in case of failure, the reactor coolant system could still perform its safety function. How? By closing the block valve. And if this valve was inoperative, the emergency core cooling system (ECCS) would automatically be activated and could easily override the coolant loss through the open PORV. At TMI the operators did not know that the PORV was stuck open, did not close the block valve, and even terminated the automatically activated ECCS.

QUESTION: Do you recall the reason that it was concluded that PORV failing in the open position was an acceptable design feature of TMI-2?

SEYFRIT ANSWER: The major one was that the high pressure safety injection system was sized to be able to provide water to the reactor at a rate greater than could be lost through the open PORV. So that there was indeed a backup system in the event of a failure. And based on the single failure criterion which has been used by the NRC traditionally, that would make it an acceptable design.

QUESTION: In other words, the assumption would be that a single failure of the PORV would not result in core uncovery because no failure with respect to ECCS was built into the analysis?

SEYFRIT ANSWER: That is correct.*

In the mid-1970s, an effort was made by the Institute of Electrical and Electronic Engineers' (IEEE) Committee on Reliability to modify the single failure criterion. The proposal to formalize probabilistic analysis as an alternative to the single failure criterion is contained in a letter dated April 19, 1976 (see Attachment 16). The committee said:

> ...we have found that some widely accepted criteria are often applied to designs in such a way that the reliability or availability of the affected system is not improved, and may even be decreased. The single failure criterion is one such criterion. There is no question that the concept of redundancy, implicit in the single failure criterion is of great importance in the design of high reliability systems. However, the criterion, as sometimes stated, has no limits, or qualifiers on the credibility of likelihood of failure or the required system reliability. The criterion is thus sometimes applied in such depth, and to such improbable situations, as to defeat the original purpose. One can indeed be led into a higher cost and lower reliability system by overemphasis on "single failure." In contrast, a probabilistic evaluation allows a balanced judgment of the reliability actually needed and allows the designer the option of defense of non-redundant components in cases where they may indeed be adequate.

This proposal was rejected by the NRC. A probabilistic approach to failure was also proposed by IEEE Standard 352-1975 (ANSI 41.4-1976, "General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems." The PORV, being an essential part in the overpressure protection system of the RCS, should have found its proper place in the spectrum of safety-related components. The FMEA proposed

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* Seyfrit deposition at 62-63, SC5

by this standard (IEEE-352) was not adopted and the issue of the PORV still remains the black-and-white issue of: to be or not be safety-related, that is the question.

In 1977, Sandia Laboratories performed a careful study of the NRC's Quality Assurance Program, reported in NUREG-0321 (reference 23). The study is very critical of the present practice utilizing the "single failure criterion" in design reviews. The study found that design measures for accommodating errors (for example, use of redundancy, design margins, etc.) in reactors are based on (1) deterministic criteria whose blanket application does not consider, for the hardware involved, factors which might affect either the failure-related or common-mode failure potentials (such as complexity, or difficulty of fabrication), or (2) a qualitative approach to reliability.

One example of the deterministic criteria approach is the "single failure criterion" used for establishing redundancy requirements. For example, three or more devices should be used if two could not be made to perform reliably enough, or if the failures tended to correlate. Design review based on unreliability rate analysis can generally identify such situations, even when there is only generic data to assess failure rates.

The latest criticism of the single failure criterion comes from the ACRS:

> The NRC should begin a study to determine if use of the single failure criterion establishes an appropriate level of reliability for reactor safety systems. Operating experience suggests that multiple failures and common-mode failures are encountered with sufficient frequency that they need more specific consideration. This study should be accompanied by concurrent consideration of how the licensing process can be modified to take account of a new set of criteria as appropriate. (reference 24).

FINDINGS AND CONCLUSIONS

The primary results of an investigation into the history and background of the PORV (pilot-operated relief valve) are presented in the following findings and conclusions:

FINDINGS

1. The PORV apparently failed in the open position at TMI-2 on March 28, 1979; TMI-2 operators had no positive indication of the open/closed position of the PORV; the absence of this signal in the control room contributed to the confusion of the operators during the TMI-2 accident.

2. Failure of the PORV in the open position results in a small-break LOCA.

3. Existing procedures did not consider a stuck-open PORV as a small-break LOCA.

4. The PORV was not classed as a safety-related component of the reactor coolant system.

5. The non-safety-related classification of the PORV can be traced to the application of the Single Failure Criterion.

6. There have been 11 failures of PWR PORVs failing in the open position before TMI-2 (March 1979) documented in this report, nine of these were in B&W plants, compared with five recognized in the NRC Summary (NUREG-0560). A number of different causes are listed for these failures.

7. Of the nine that failed in the open position in B&W plants, B&W was aware of seven. Of the 11 that failed in the open position, 8 were supplied by one supplier, which was aware of the 4 supplied by them that were contained in the NRC Summary (NUREG-0560).

8. Failure analyses and corrective actions following these PORV failures were insufficient and ineffective.

9. The NRC has not highlighted PORV problems as an unresolved safety issue, as an abnormal occurrence, or as a generic problem.

10. Standards for PORV design, testing, and function performance are not available.

11. National reliability data systems are only in the early stages of development by both EPRI and the NRC.

CONCLUSION

The TMI-2 accident would probably not have progressed beyond a severe feedwater transient, had the PORV been recognized and treated as a safety-related component.

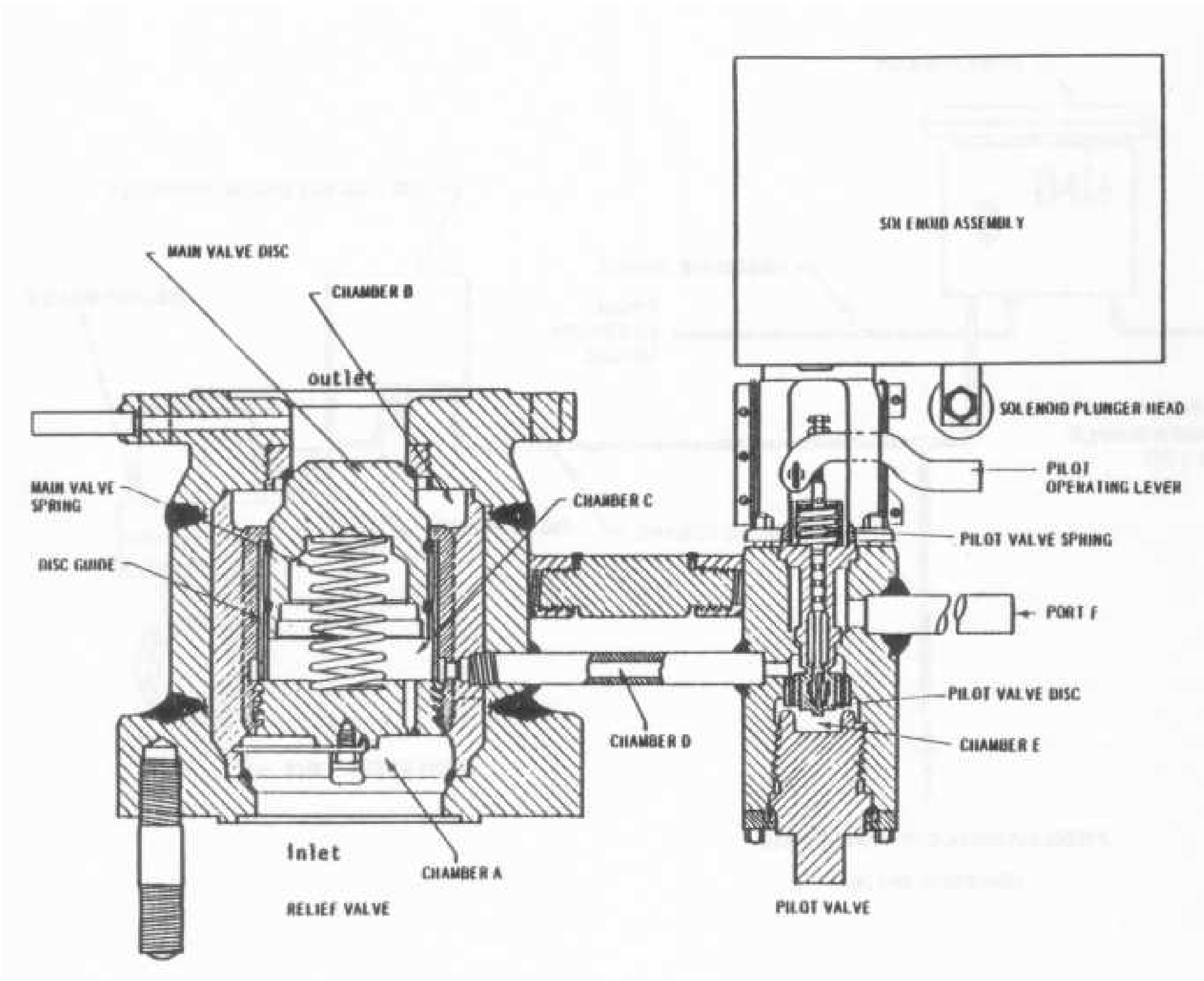
ACRONYMS

AEC	Atomic Energy Commission
ACRS	Advisory Committee for Reactor Safeguards
A/E	Architect Engineer
ANS	American Nuclear Society
ANSI	American National Standards Institute
APPA	American Public Power Association
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
B&W	Babcock and Wilcox
BOP	Balance of Plant
BPV	Boiler and Pressure Vessel Code
B&R	Burns and Roe
CFR	Code of Federal Regulations
C-E	Combustion Engineering
DOE	Department of Energy
EEI	Edison Electric Institute
EPRI	Electric Power Research Institute (of EEI and APPA)
FIRR	Failure and Incidents Reports Review Committee (of the NSMB)
FMEA	Failure Mode and Effects Analysis (IEEE Std. 352)
FSAR	Final Safety Analysis Report
GDC	General Design Criteria (Appendix A of 10 CFR 50)
GPUSC	General Public Utilities Service Corporation
GPU	General Public Utilities
IEEE	Institute of Electrical and Electronic Engineers
JCAE	Joint Committee on Atomic Energy
NNI	Non-Nuclear Instrumentation
NPEX	Nuclear Power Engineering Committee (of the IEEE/PES)
NPRD	Nuclear Power Reliability Data System (operated by Southwest Res.)
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulations (and licensing)
NSMB	Nuclear Standards Management Board
NSSS	Nuclear Steam Supply System
NUREG	NRC Publication Code Name
PARA	Problem Analysis and Recommended Action (of FIRR)
PES	Power Engineering Society (of the IEEE)
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
Tech Specs	Technical Specifications
SRP	Standard Review Plant
SRS	System Requirements Specifications
SRV	Safety Relief Valve

Definitions

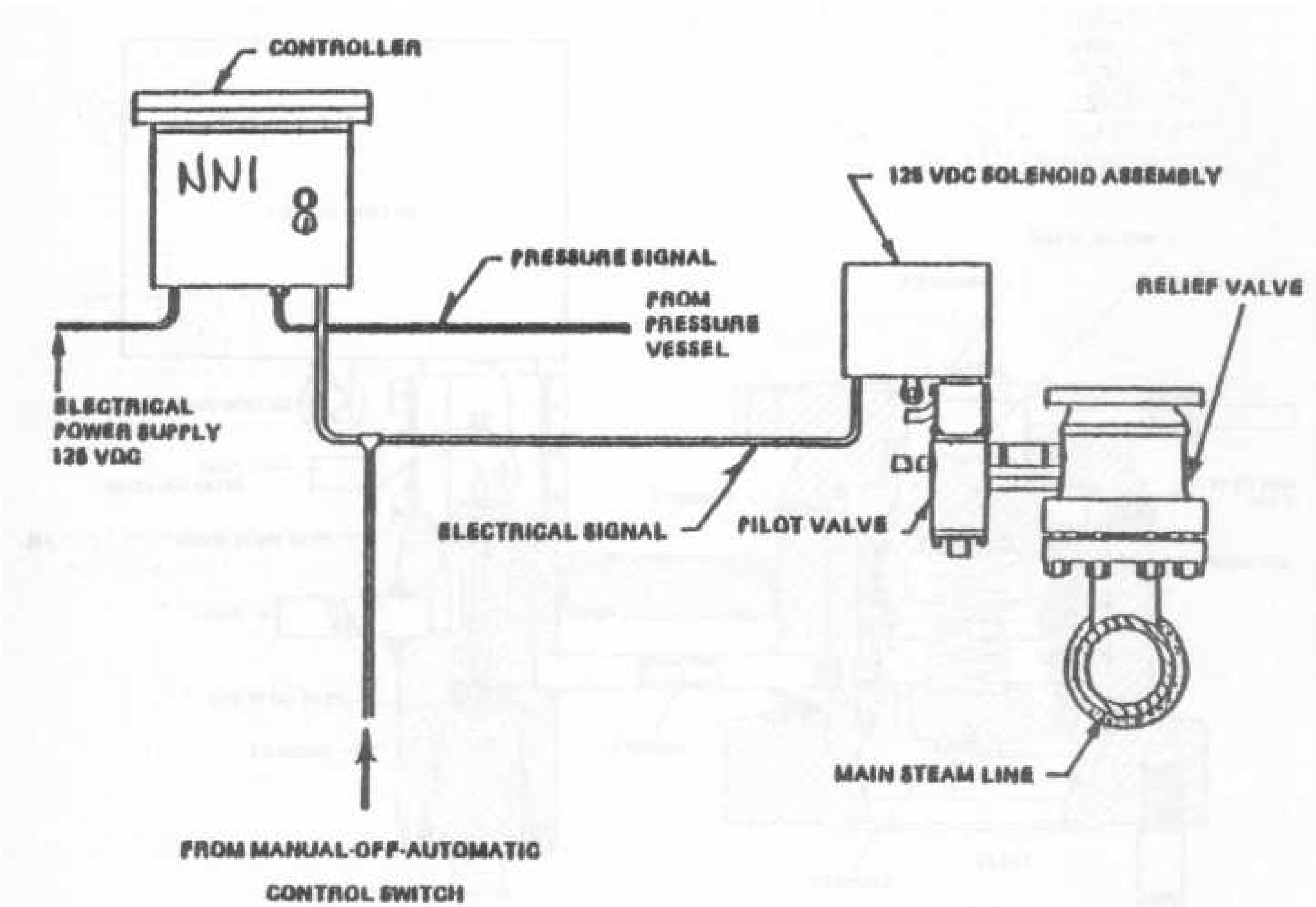
ANSI B95.1-1977, Terminology for Pressure Relief Devices

FIGURE 1: Dresser Pilot-Operated Relief Valve



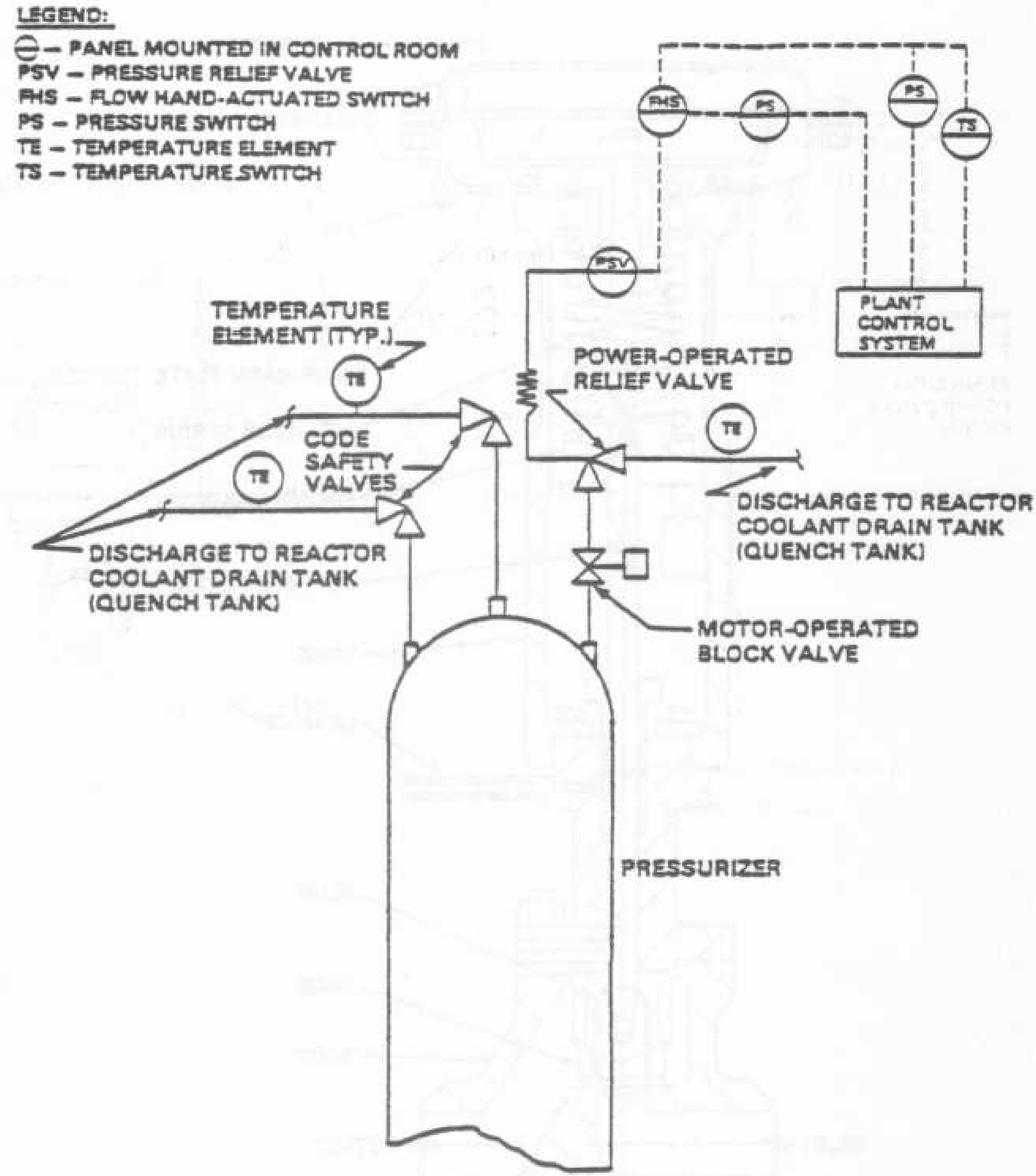
Source: NUREG-0462, "Technical Report on Operating Experience with BWR Pressure Relief Systems," NRC, July 1978.

FIGURE 2: Dresser Pilot-Operated Relief Valve and Controls



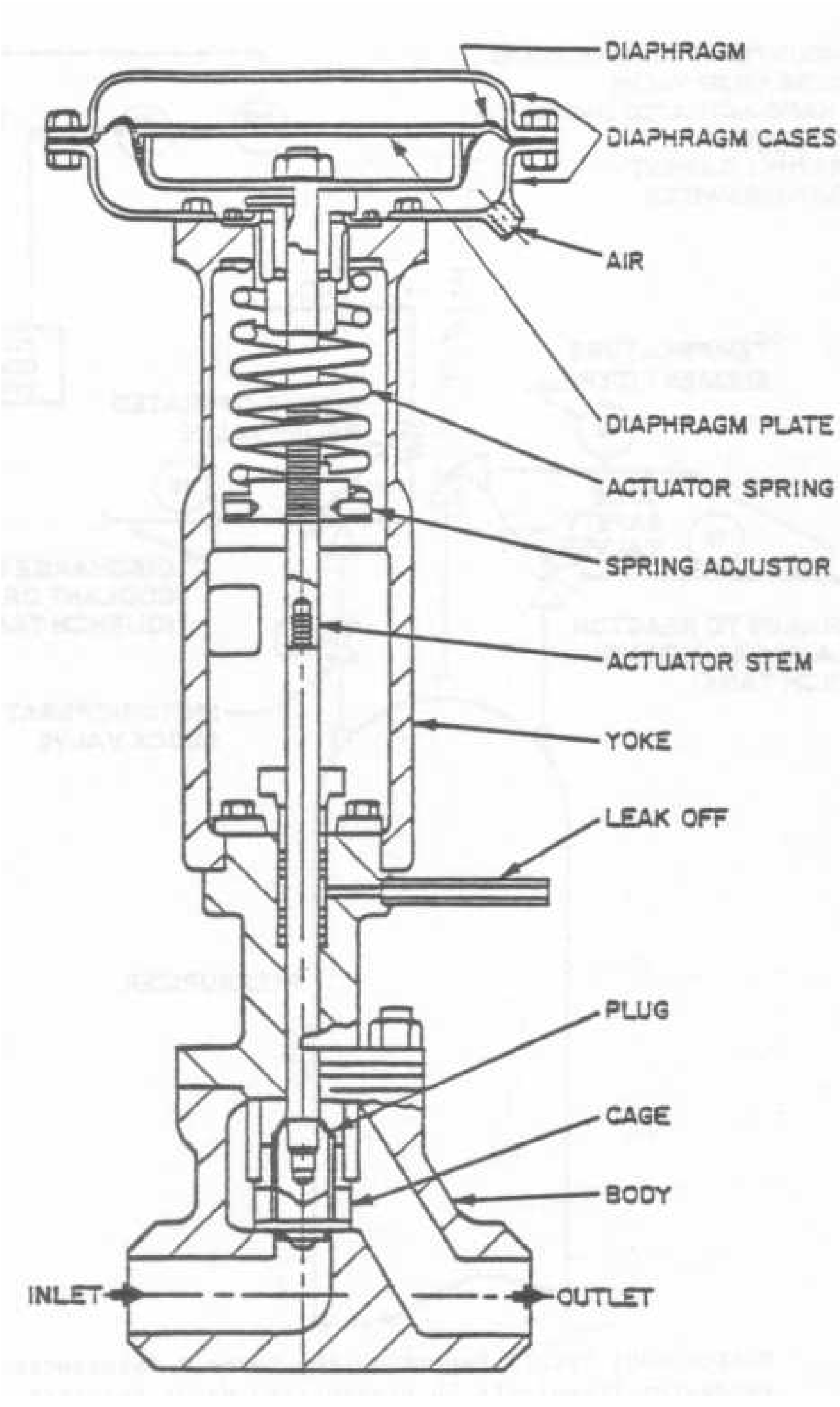
Source: Dresser Industries.

FIGURE 3: Typical Arrangement of Relief and Safety Valves on Pressurizer (B&W)



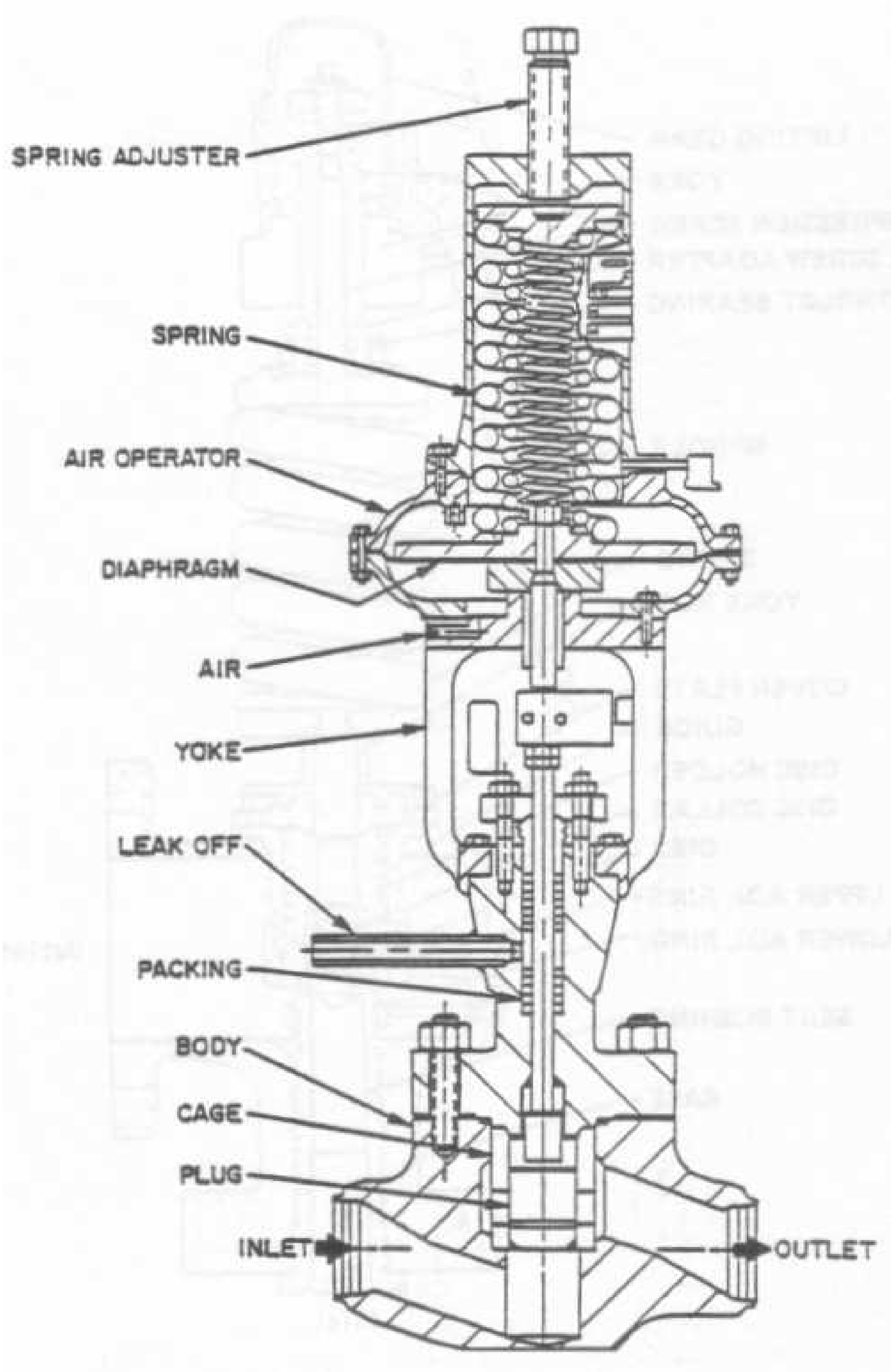
Source: NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company" (Tedesco Report), NRC, May 1979.

FIGURE 4: Westinghouse Air-Operated Globe PORV -- Type 1



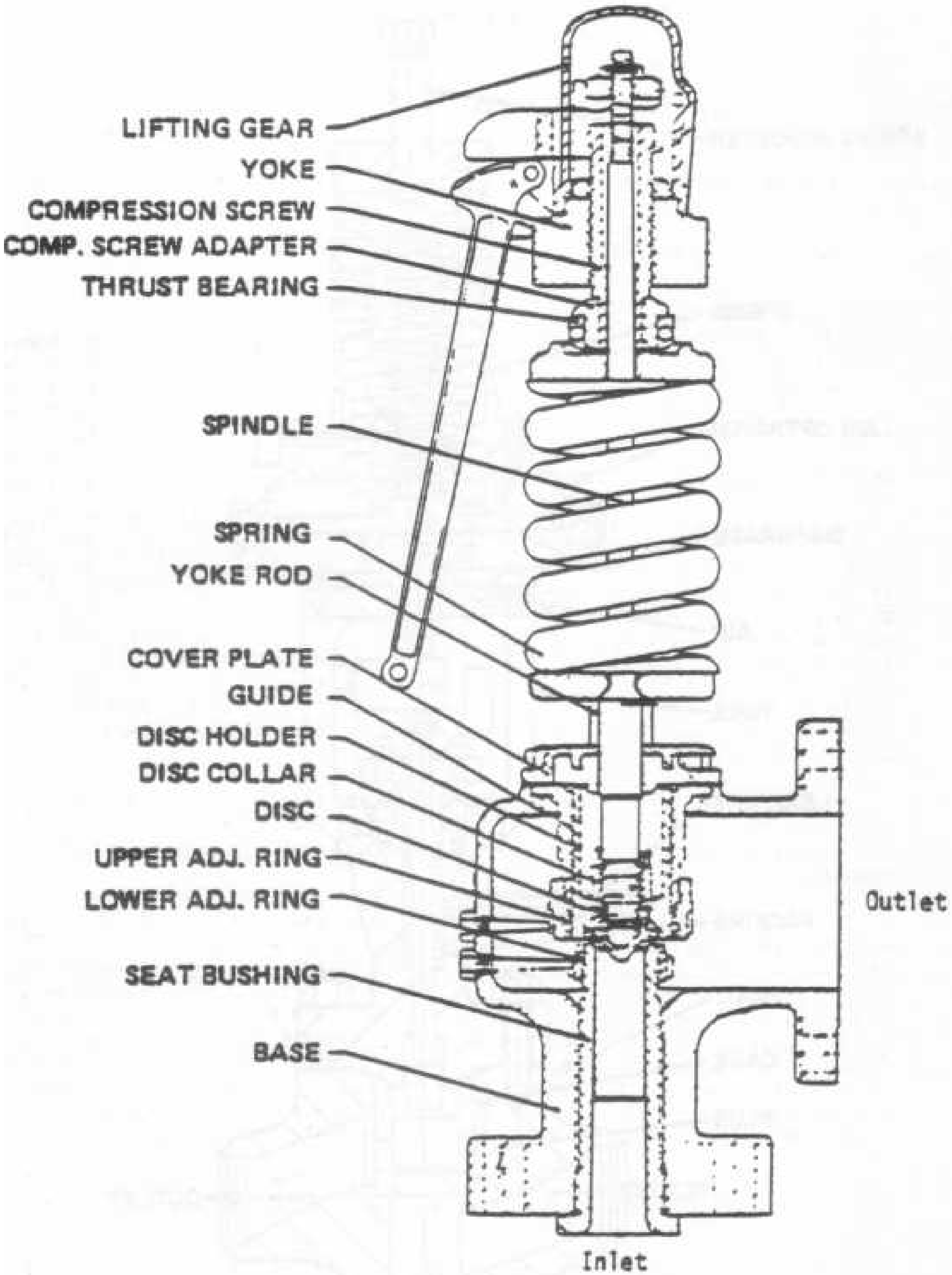
Source: Westinghouse Corporation.

FIGURE 5: Westinghouse Air-Operated Globe PORV -- Type 2



Source: Westinghouse Corporation.

FIGURE 6: Safety Valve



Source: NUREG-0462, "Technical Report on Operating Experience with BWR Pressure Relief Systems," NRC, July 1978.

ATTACHMENT 1

COMMITTEE CORRESPONDENCE

ganization: American National Standards Committee

Committee: B16 Subcommittee H

Subject: ANSI N41.6 (IEEE 382)

Address Writer Care of:
Pacific Gas and Electric Company
77 Beale Street
San Francisco, California 94106

Date: July 24, 1979

To: Mr. Melvin R. Green
American Society of Mechanical Engineers
345 East 47th Street
New York, New York 10017

Copies to: Messrs. A. Bagner
W. G. Canham
D. K. Greenwald
B. J. Milleville
M. A. Moler
R. G. Visalli
R. V. Warrick

Dear Mel:

This letter is to bring you up to date regarding the status of B16 S/C H activities regarding the standard on qualification of valve actuators, ANSI N41.6 (IEEE 382). As you will recall, there are presently two versions of this standard. One has been prepared by a Joint Committee of members from B16 S/C H, Task Force 1, and IEEE Task Force 382. This version has been successfully balloted by IEEE (NPEC S/C 2 (qualification) as well as NPEC itself) and, as I understand, is on the agenda for the IEEE Standards Board meeting in September, 1979. This version was also balloted and disapproved by B16 S/C H. A second version has been prepared by an Ad Hoc Committee of B16 S/C H and reviewed at the March meeting of S/C H. This version, if balloted by S/C H, would probably be approved.

I called you on April 2, 1979, and discussed this matter with you. You asked Bert Stanley of the IEEE Staff to call me, which he did on April 3. The matter was also discussed with Marv Moler, Vice Chairman of the Joint Committee, who also talked to Bert Stanley.

A meeting was suggested between the Joint Committee and a few members of an Ad Hoc Committee from B16 S/C H to try and resolve the differences in the two versions of the standard. There are differences both in format and substance. I understand Bert Stanley requested that the Joint Committee be polled to see if they were willing to meet with the B16 S/C H Ad Hoc Committee and the results of this poll were negative. Because of this, we apparently have an impasse between IEEE and B16 S/C H wherein the Joint Committee, IEEE's S/C 2 of NPEC and the NPEC wish the Joint Committee's version to be approved and published. B16 S/C H, on the other hand, has problems with the Joint Committee version and would like to try and resolve the problems.

The resolution of this matter may have some bearing on the final approval of the ANSI B16.41 Standard dealing with qualification of valve assemblies.

Please let me know if there is anything further I can do to help resolve this apparent impasse.

Sincerely,

J. O. Schuyler

J. O. Schuyler
Chairman

ATTACHMENT 2

President's Commission
on the Accident at Three Mile Island
2100 M Street, NW Washington, DC 20037

SEP 2 0 1979

MEMORANDUM FOR THE RECORD

SUBJECT: Visit to Dresser Industries Plant, Alexandria, Louisiana

On September 19, 1979 the undersigned accompanied by Mr. Art Carr, consultant, visited the Alexandria, Louisiana,. plant of Dresser Industries, the suppliers of the PORV (pilot operated relief valve) installed in 1111-2 at the time of the March 28, 1979 accident. The principal contacts during the visit were Mr. William Tacy, Jr., Manager Product Engineering and Mr. John Richardson. Manager Nuclear Operations.

The purpose of this memorandum is to note initial observations and impressions. Further analysis of information requested to be supplied by Dresser (letter of 9%20%79), nay result in further elaboration of these observations and impressions.

1. Dresser was aware of PORV problems contained in NUREG 00560 (when it was publised), their main sources of information has been customer reports, Dresser field service representatives. and LERs.

2. Dresser was not aware of any PORV problems other than those contained in NUREG 00560.

3. The PORV installed in TMI-2 at time of the accident was delivered in 1972; its history includes earlier problems when installed in TMI-l; .details to be furnished by Dresser.

4. The PORV is a design that has been produced for about 15 years; 628 units of the same design have been produced for non nuclear plants, 31 units have been produced for nuclear plants.

5. The main differences in PORV's produced for nuclear plants over the non nuclear plants is harder valve disc and valve seat materials, Quality Assurance Program Controls and Tracea bility.

6. Only two design improvements have been incorporated in the design; retrofitting has been done. The design changes were to hinge pin of lever to pilot valve as results of failure involving buildup of boric acid crystals; and a production simplification to the base of the valve body.

7. FMEA's (failure modes and effects analyses) not normally accomplished on Dresser valve designs. Such an analysis was done after the March 28, 1979 accident at TMI-2 to postulate failure modes causes for stuck open disc (if that turns out to be the TMI-2 PORV problem). The postulated causes are:

   a. Electrical pulses to the solenoid `to cause disc rapid opening-and closing that could damage seat: or disc.
   b. Piston rim-,gal l
   c. Foreign material entrapment

8. Dresser suggested that there are ways to provide maintenance for leaky PORY while plant **is in** operation, additional **details** expected.

9. Dresser noted that they have been subjected to numerous audits' by AFC/NRC, ASME, Hartford Boiler Insurance (an independent auditor required by ASME), B&W and perhaps GPU.

10. The Nuclear Operations Quality Assurance Manual,, against which the audits are performed, appeared to be complete document; it **will be** examined more closely .when a copy is obtained. The current version of the manual was.noted by Dresser, as containing much more detail than the manual in effect when the TMI-2 PORV was prod+ured.

11. Code Safety **valve, similar** to type on the TMI-z pressurizer was observed undergoing final steam testings and verification of set points. The "pass critiera" was three consecutive actuations within spedified pressure limits. It was. noted by Dresser that such actuations did not require valve rework.

12. General plant practices seemed normal for this type of metal handling, fabrication and machining operation and **inspection** effort was evident; an **additional** look into the controls to provide contamination-free finished valves will be accomplished when the Quality Assurance Manual is available.

William Bland

:: V. Johnson
L. Jaffe
D. Reilly
A. Carr
S. Gorinson

ATTACHMENT 3

President's Commission on the Accident at Three Mile Island
2100 M Street, NW Washington, DC 20037

MEMORANDUM FOR RECORD September 25, 1979

SUBJECT: Visit to Dresser Industries - Alexandria, Louisiana

Dresser Industries, Alexandria, Louisiana was visited September 19, 1979 by the undersigned and Mr. William Bland. Personnel contacted at Dresser were: Mr. William Tacy, Manager Product Engineering, and Mr. John Richardson, Manager Nuclear Operations. The pilot operated relief valve (PORV) that is installed in TMI-2 was manufactured at this plant and delivered in February 1972.

Information, observations and impressions obtained during this visit are:

> The design of the PORV used at TMI has been in production for approximately 15 years. A total of 628 of these valves have been delivered to commercial users and 31 to nuclear customers.
>
> The company is only aware of one significant nuclear valve malfunction. This was caused by a buildup of boric acid crystals on the hinge pin of the lever to the pilot valve. As a result, the hinge pin material was changed and all valves in the field were modified. They have had about six cases of damaged seats on commercial valves during the past few years due to chatter caused by the control circuit.
>
> Dresser does not normally perform failure mode and effects analysis (FMEA) or establish failure rates for their valves. They did develop some failure modes for the PORV after the TMI incident. These modes and there probability of occurrence are:

Chatter due from control circuit	50% probability
Galling between piston ring and guide	5% probability
Foreign material on main seat - full open	5% probability
- part open	50% probability

Dresser has a quality assurance manual that has been approved by an ASME inspector. The manual appears to describe a comprehensive quality system. A copy has been requested and will be reviewed later.

The Dresser quality system is audited quite often. NRC audits are conducted about once a year, ASME every three years and there are numerous customer audits.

Based on a brief tour of the plant, it was concluded that the company has the capability to produce quality valves.

Two PORV's were observed in the shipping area. They were to be delivered to Combustion Engineering. It was stated that these two valves were "N" stamped. This means that they were built to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. A comparison of this code with NRC requirements is attached.

Arthur M Carr

Arthur M. Carr

Attachment

cc:
W. Bland
D. Reilly

Summary of Section III of ASME Boiler and Pressure Vessel Code Requirements for Class I, 2 and 3 Components and Quality Group A, B, C and D Classifications of NRC Regulatory Guide 1. 26

ASME Boiler and Pressure Vessel Code

Section III

Section III contains requirements for the construction of nuclear power plant items such as vessels, storage tanks, piping systems, pumps, valves, and core support structure, and component supports for use in, or containment of, portions of the nuclear power system of any power plant.

The code recognizes the different levels of importance associated with the function of each item as related to the safe operation of the plant. Construction rules for components are specified as code class 1, 2 and 3. The code does not provide guidance in the selection of a specific classification to fit a component in a given system. This guidance is found in Section 50. 55a of 10CFR Part 50 and NRC Regulatory Guide 1. 26.

Design, fabrication, installation, testing and quality requirements are specified for each class. Class 1 requirements are the most stringent and class 3 the least stringent. A large portion of the requirements are the same for all classes. Examples of differences in requirements are: ultrasonic testing is required for class 1 forgings but not class 2 or 3. Detailed (6 pages) quality assurance requirements are specified for class 1 and 2 while rather general (2 pages) quality control requirements are specified for class 3.

Title 10 Code of Federal Regulations Part 50. 55a

10 CFR 50. 55a, "Codes and Standards, " requires that components that are a part of the reactor coolant pressure boundary meet the requirements for class 1 of section III of the ASME Boiler and Pressure Vessel Code.

Footnote 2 of that section excludes components from this requirement if, after a failure of the component, the reactor can be shutdown and cooled down in an orderly manner, assuming makeup is provided by the reactor makeup system only.

NRC Regulatory Guide 1. 26, "Quality Group Classification and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants"

This document provides guidance for classification of components that are not ASME class 1 (note - 10 CFR 50. 55a provides guidance for class 1). It defines four quality groups, A through D.

Group A corresponds to ASME class 1 and is defined in 10 CFR 50. 55a.

Group corresponds to ASME class 2 and applies to components that are either a part of the reactor coolant pressure boundary but excluded from the requirements of ASME class 1 by footnote 2 of 10 CFR 50. 55a, or not a part of the reactor coolant pressure boundary but part of:

Emergency core cooling system

Postaccident containment heat removal system

Postaccident fission product removal system

The steam system of moiling water reactors from the containment isolation valve to the turbine stop and bypass valves and connecting piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation

Portions of the steam and feedwater system of pressurized water systems from and including the secondary side of steam generators up to and including the outer most containment isolation valves and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation

Systems connected to the reactor coolant pressure boundary and not isolated by two valves, which are normally closed or capable of automatic closure

Group C corresponds to ASME class 3 and applies to components that are not a part of the reactor coolant pressure boundary or included in group B but part of:

Cooling water and auxiliary feedwater systems designed for (1) emergency core cooling, (2) postaccident containment heat removal, (3) postaccident containment atmosphere cleanup or (4) residual heat removal from the reactor and from the spent fuel storage pool

Cooling water and seal water systems for components such as reactor coolant pumps, diesels and control room

Systems connected to the reactor coolant pressure boundary and isolated by two valves

Systems other than radioactive waste management systems not covered above that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 0. 5 rem to the whole body or its equivalent to any part of the body

Group D applies to components not part of the reactor coolant pressure boundary or included in groups B or C but part of systems that contain or may contain radioactive material.

Stamping

Section III of the ASME Boiler and Pressure Vessel Code describes stamping requirements for components built to the requirements of Section III. Class I components are to be stamped "N-1", class 2 or 3 components are to be stamped "N-2" or "N-3. " It is stated in 10 CFR 50. 55a that "ASME code N-symbol need not be applied. " This statement is also contained in NRC Regulatory Guide 1. 26.

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION
LYNCHBURG, VIRGINIA

INSPECTION REPORT

SUPPLIER Dresser Industries INSP. DATE March 13 & 14, 1973
ADDRESS Alexandria, La. REPORT # 92
B&W P.O.# 02266CLS ITEM # PAGE # OF 2
COMPONENT Pressurizer Relief Valve CUSTOMER Jersey Central
B&W MK. # RC-RV1A, RC-RV1b CONTRACT # 620-0006
SUPPLIER SER. # BN04231, BN04232 CUSTOMER TAG #
SOURCE INSP. PLAN #

1. TYPE OF INSPECTION
Final assembly functional test
Wall thickness minimum dimensions

2. PERSONNEL CONTACTED G. J. Harrison, Sales Engineer, Gulf X Ray.
B. F. Brunson, Chief Inspector, Dresser Industries
T. Barnes, Inspector, Dresser Industries
E. E. Harris, Q.A. Tech. Ass't, Dresser Industries

3. DRAWINGS AND/OR SPECIFICATIONS USED IN MAKING INSPECTION
Dresser 05197 Rev. 4

4. INSPECTION SCHEDULE AND COMPLETION DATE
Shipping date approx. March 20, 1973

5. RESULTS OF INSPECTION
Satisfactory

6. ALL INSPECTIONS ON THIS ORDER COMPLETED [X] (check if yes)

DISTRIBUTION:

P. O. FILE INSPECTED BY C. M. Haney
VENDOR FILE MGR. Q.C.S. C M [signature]
PURCHASING (BUYER) G. T. Card DATE 3-27-73
Q.A. ENGINEERING E. P. Fair
COG. ENGR. J. D. Dempsey
PROJ. MGMT. W. A. Cobb

ATTACHMENT 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, 0. C. 20555

SEP 2 6 1978

NOTE TO: W. F. Anderson, SD
L. Porse, SD
J. Zudans, DOR

FROM: R. Bosnak, DSS

SUBJECT: INSERVICE TESTING OF VALVES

In reviews of inservice testing programs for valves, a question has arisen concerning the Winter '77 change to IWV-1100. We understand the words "of'preventing the consequences of an accident" were left out because of the Code's reluctance to deal with systems safety considerations. Many safety and relief valves in Class 2 and 3 systems are not specifically used in shutdown or mitigating the consequences of an accident, but if they failed under overpressure conditions, could result'in a system failure. The Winter '77 Addenda could be read to literally exclude such valves from the inservice testing program. We do not believe that this was the intent and suggest a modification as shown on the attached page. We recommend this be handled via normal Code Committee procedures **in** S/C XI and the S/G on Testing of Pumps and Valves.

R. **Bosnak,** Chief
Mechanical Engineering Branch
Division of Systems Safety

Attachment:
Art. IWV-1000

cc w/att.:
J. Knight, DSS
H. Brammer, DSS
F_ Cherny, DSS
Ibonan, DOR
'fmvacs, DSS

ATTACHMENT 6

THE BEZNAU, SWITZERLAND, INCIDENT, AUGUST 20,1974

On Aug. 20, 1974, a turbine tripped at the NOK-1 nuclear facility in Beznau, and eventually the two PORVs on this plant opened to relieve excess pressure. One PORV closed and the other stuck open. While pressure fell, pressurizer level rose and remained off-scale high for 3 to 5 minutes. Although steam bubbles were forming in the primary system, the HPI, which actuated only if pressure and -- note: and -- level were trending together, did not come on.

The failure to close was detected in a few minutes by the operators, who immediately isolated the valve by closing the block valve in series with the PORV. This action terminated the incident. The failure to reclose was due to the rupture of the cast-iron yoke (frame) between the valve operator and the valve body.

Pressurizer level fell rapidly, and -- at about 12 minutes into the event -- the HPI was automatically actuated by both low pressure and low level. Had the Beznau operator failed to shut the block valve, the consequences of the Beznau transient might have equaled those of the TMI accident. The difference would have been in the cause -- not in operator error in terminating HPI as at TMI, but in a design that prevented HPI from automatically actuating.

The NRC was not told about Beznau by either Westinghouse or the Swiss Federal Office of Energy until after the TMI-2 accident.* No change was made as a result of the Beznau transient in the 23 Westinghouse reactors operating in the United States that also had coincident logic actuation of HPI.'^ Instead, these plants utilized coincident logic until the TMI-2 accident and the resulting actions by the NRC and Westinghouse.

---

* LaFleur deposition at 55, 82.
Thadani deposition at 71.

ATTACHMENT 7

PALISADES INCIDENT, SEPTEMBER 8, 1971*

On Sept. 8, at 1:35 p.m., a technician de-energized the breakers to the reactor protective system to install a minor modification. This de-energized the feed to the electromatic relief valve solenoid allowing the valves to open.

The primary system pressure decreased to a low point of approximately 1,280 psia over a period of 2 to 3 minutes until the blowdown was terminated by closure of the motor-operated block valve. The system pressure and temperature were back to normal in approximately one hour. The system was in a hot shutdown condition.

The basic cause of this incident was the non-standard designation of contacts. The technician was mislead by the 'a' contact designation as shown on the architect/engineer drawing when in fact the circuit is wired using 'b' contact(s). This led him to believe the relief valves would not open when the power was removed (by opening the breaker).

A review of the control scheme design was conducted by Combustion-Engineering and the utility. Corrective action documentation is not available.

This incident is not reported on any of the NRC precursor lists. Consumers Power's corrective action eliminated the PORV from the RCS by closing the block valve for good. The PORV in this case is not necessary to protect the safety valves from popping open. In the unlikely event of a severe transient, the reactor would trip and further pressure increase would lift the safety valves.

Combustion-Engineering modified instructions so as to have the set point for the PORV coincide with reactor trips, both using the same signal. In their current design, the PORV is eliminated altogether. There are four safety code valves on the pressurizer (reference 14).

---

* From an abnormal occurrence report to AEC, dated Sept. 16, 1971, from Robert L. Haueter, Palisades, production superintendent, nuclear, to Dr. Peter Morris, director, Division of Reactor Licensing.

ATTACHMENT 8

ARKANSAS NUCLEAR ONE-1 INCIDENT, BEFORE SEPTEMBER 19, 1974

ANO-1 achieved full power Dec. 8, 1974. During startup testing in late August or early September, the PORV lifted and stuck open. The fault was recognized and diagnosed by Arkansas Power and Light Company (APLC) and a field design change request was processed Sept. 19, 1974, to change the location of the pilot vent line. The 1/2-inch vent line from the pilot should eventually lead to the quench tank. B&W's installation instructions "recommended that the pilot vent into the discharge piping downstream from the main valve" (i.e. the PORV). Pipe installation and their field changes are the responsibility of the architect engineer -- Bechtel in this case. B&W diligently searched all their filed reports and startup records and could not find a single mention of the stuck-open PORV or of the corrective action in connection with the pilot vent pipe.

This precursor event is mentioned in NUREG 0560 (reference 4), and came to light at the NRC, in a response from APLC to the I&E Bulletin 79-05A. It is not in the LER file. B&W record shows that the plant was down from Sept. 13 to 19, 1974. At that time, the B&W record indicates that Bechtel determined that the snubbers on the PORV discharge line were too light. Heavier snubbers were installed.

It is not known whether the generic issue is recognized. At issue is the location where the 1/2-inch vent line joins the 4-inch discharge pipe. It must be far enough downstream so as to sense a sufficient pressure drop in order to close the pilot. If the pressure drop is insufficient to activate the pilot, the pilot in turn will keep the PORV from closing as required.

Conflicting and confusing information from utility, NRC, and B&W. No LER processed. Field change not recorded at the NSSS supplier. Possibility of a generic problem not considered.

ATTACHMENT 9

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To: D. H. Roy	
From: G. T. Fairburn; Service Manager	BDS 663.
Cust.	File No. or Ref.
Subj. Pilot-Operated Relief Valve	Date May 9, 1979

This letter to cover one customer and one subject only.

Per your request to Mr. G. M. Olds the Service Managers have contacted each of our operating customers concerning the opening or failing open of the PORV on loss of power either to the valve proper or to the electrical circuitry which controls the valve. We were given the following responses:

SMUD	- fails shut
Oconee	- does not open
Davis-Besse	- does not open
ANO	- if closed, will not open; if open, may not close
TMI-1	- if closed, remains closed; if open, remains open on loss of power to control circuitry
FPC	- fails shut

It is not apparent that even if the valve does stay open that it is completely bad. It provides a relief path thus preventing the opening of the safety valves and can be controlled by the use of the block or isolation valve.

G T Fairburn

cc: G.M.Olds
E.A.Womack
G.J.Brazill
K.E.Suhrke
C.D.Russell
R.C.Luker
J.T.Janis
K.R.Ellison

ATTACHMENT 10

..E MILE ISLAND NUCLEAR STATION

GPU STARTUP PROBLEM REPORT

GPU NUMBER 2718

ORGANIZATION SERIAL NUMBER

TMI UNIT 2

SYSTEM: RCS

TP NO. MTX NO. 147

PROBLEM DESCRIPTION: The electromatic relief valve opens on loss of power to its control bistable. Suggest changing this or providing an indication on Control Panel that indicates valve has an open signal

BY: [illegible] Tole

ORGANIZATION: GPU

DATE: 3-30-78

FOR RESOLUTION BY: B&W - Rogers DATE SENT: 3-30-78

ROPOSED RESOLUTION:

SEE ATTACHED

BY: [illegible]

DATE: 4-3-78

FOR ACTION BY: B&R - Bronnewell DATE SENT: 4-3-78

Pri 1 → issue ECM to accomplish

rIroroSKI)rtr~~:or,rrriol-r

h&W ha:, reviewed the clectromatic relief valve logic and agrees to the concept of leaving rclicf valve fail closed on loam of 11111 power supply to the Ili-Lour Monitor (3-10-12). To achieve this condition, ;:witch g-1 should be in the dcencrgized mode and the wirinc modification be made as indicated in the attached sketch. Per your requcst,a formal field chance will follow.

To provide an indication that the electromatic relief valve has an open signal, a review of the ccnstruction schematics indicates that a control room indicating light operated from power to the solenoid can be added without additional cabling. (Refer to R&R drawing ,13079, sheet lie.) This light could be actuated by the same auxiliary relay in the power distribution panel that supplies power to the valve solenoid.

MEMORANDUM

ATTACHMENT 11

BURNS and ROE, Inc.

DATE 8/1/79

DEPARTMENT

AUG 22 1979

TO W. R. Cobean, Jr.

FROM R. S. Gagliardo

SUBJECT W.O. 3475
Jersey Central Power & Light Company
Three Mile Island Nuclear Station Unit #2
Electromatic Relief Valve RC-R2 Controls

Memo, W. R. Cobean, Jr. to R. S. Gagliardo, dated 7/25/79

COPIES TO:

ASDam
KBrockwell
FASpanganberg
HRLane
MCrane, w/att.
KABogumirskas, w/att.
RSGagliardo, w/a
pf (3)
db

Per your request, we have investigated the history of the controls for the Electromatic Relief Valve (EMRV), RC-R2, (B&W Tag. No. RC-RV2) and have determined the following:

1. The control circuit for RC-R2 was designed by B&R in accordance with B&W schematic diagram 113658C (Att. #4). The B&R elementary diagram number is 3079 sh. 14.

   As originally designed, RC-R2 closed upon deenergization of the DC pilot solenoid due to the opening of a control device contact (i.e., control switch or bistable) or due to the loss of DC control power. For example, placing the control switch in the "OFF" position would deenergize relay RX, a contact of which would deenergize the pilot solenoid PA. This would cause the EMRV to close.

   The bistable (RC3-PS8) is located in one of the NNI cabinets supplied by B&W. Loss of power to the bistable would result in a closed contact in the automatic control circuit for the pilot solenoid which would energize the solenoid and cause the EMRV to open. This operation is shown on drawing 3079, sh. 14, Rev. 11 (Att. #6), which reflects the post-TMI task force state of the circuit.

   In summary, loss of DC power to the pilot solenoid would not cause the electromatic relief valve to open. However, loss of AC power to the NNI system bistable could cause the valve to open if DC control power was available and the control switch was in the "AUTO" position.

2. The EM BV controls were revised by FCR 35, Rev. 2 (B&W FC 34, Rev. 1) to incorporate the addition of the NDTT interlocks. (Att. #7). This revision did not change the operation of the EM RV on loss of DC' control power.

3. GPU Problem Report 2718 (Att. #8) was issued on 3/30/78 to B&W (Rodgers) requesting that the control circuit for the EMRV be modified so that the valve would not open on loss of power to the NNI bistable (RC3-PS8). GPU also requested indication in the Control Room of an open signal to the valve if modification of the control circuit was not possible.

   B&W concurred with the GPU requests and recommended wiring changes to the NNI bistable (RC3-PS8) to modify the control circuit. B&W also agreed to provide a red light on Panel 4 to indicate that the pilot solenoid was receiving an open signal.

4. PR 2718 was assigned to B&R on 4/3/78, requesting that an ECM be issued' to implement the B&W recommendations. ECM 5934 (Att. #9) was prepared and issued on 4/6/78. The ECM added the "Open Signal On" indicating light.on'Panel 4 and made the required modifications to the NNI cabinets to change the failure mode of the RC3-PS8 bistable. In addition, modifications were made to the NNI cabinets to change the failure modes of the RC3A-PS9 and RCSA-TS1 bistables used in the NDTT circuitry. The bistables were all modified so that loss of power to them would not cause the ARV to open.

5. GPU Problem Report 2731 (Att. #10) was issued on 4/1Q/78 requesting that the failure mode for bistables RC3A-PS4 and.RC5A-TS1 not be changed as contained in ECM 5934.

6. B&R prepared and issued ECM 5934, Rev. 1 (Att. #11) on 4/14/78, recinding the changes to bistables RC3A-PS9 and RC5A-TS1 contained in the initial issue of the ECM.

7. The control circuit incorporating all of the above changes is shown on elementary 3079, sh. 14, Rev. 13 (Att. #12). This drawing reflects the circuitry as it existed on March 28, 1979.

The enclosed "Summary of Attachments" provides an index of the pertinent documents that determined the controls for the Electromatic Relief Valve.

R. S. Gagliardo
R. S. Gagliardo

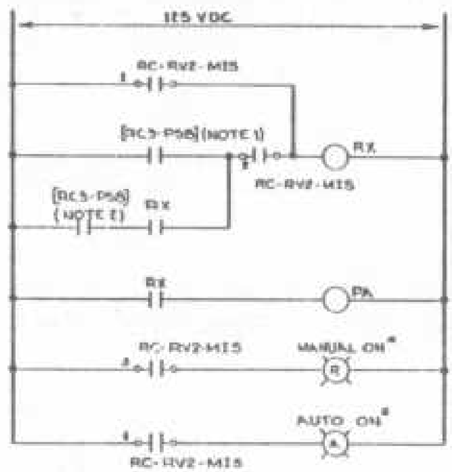
RSG/sjm

attachments

ATTACHMENT 12

REVISIONS

	DESCRIPTION	DATE	APPROVAL
1	ADDED RC-RV2-MIS SWITCH PER P&I 2161 5F-REV. 4. AND SC-91. REF DWG. NO. 1 CHANGED FROM 2161 7F TO 2161 5F PER P&I DWG. 2161 5F-REV. 1		



COMPONENT NO.
RC-RV2

RC-RV2-MIS

CONTACT	ON	OFF	AUTO
1	X		
2			X
3	X		
4			X

LEGEND

PA - PILOT ACTUATOR

NOTES:

1- HIGH SET POINT

2- LOW SET POINT

JERSEY CENTRAL POWER & LIGHT COMPANY

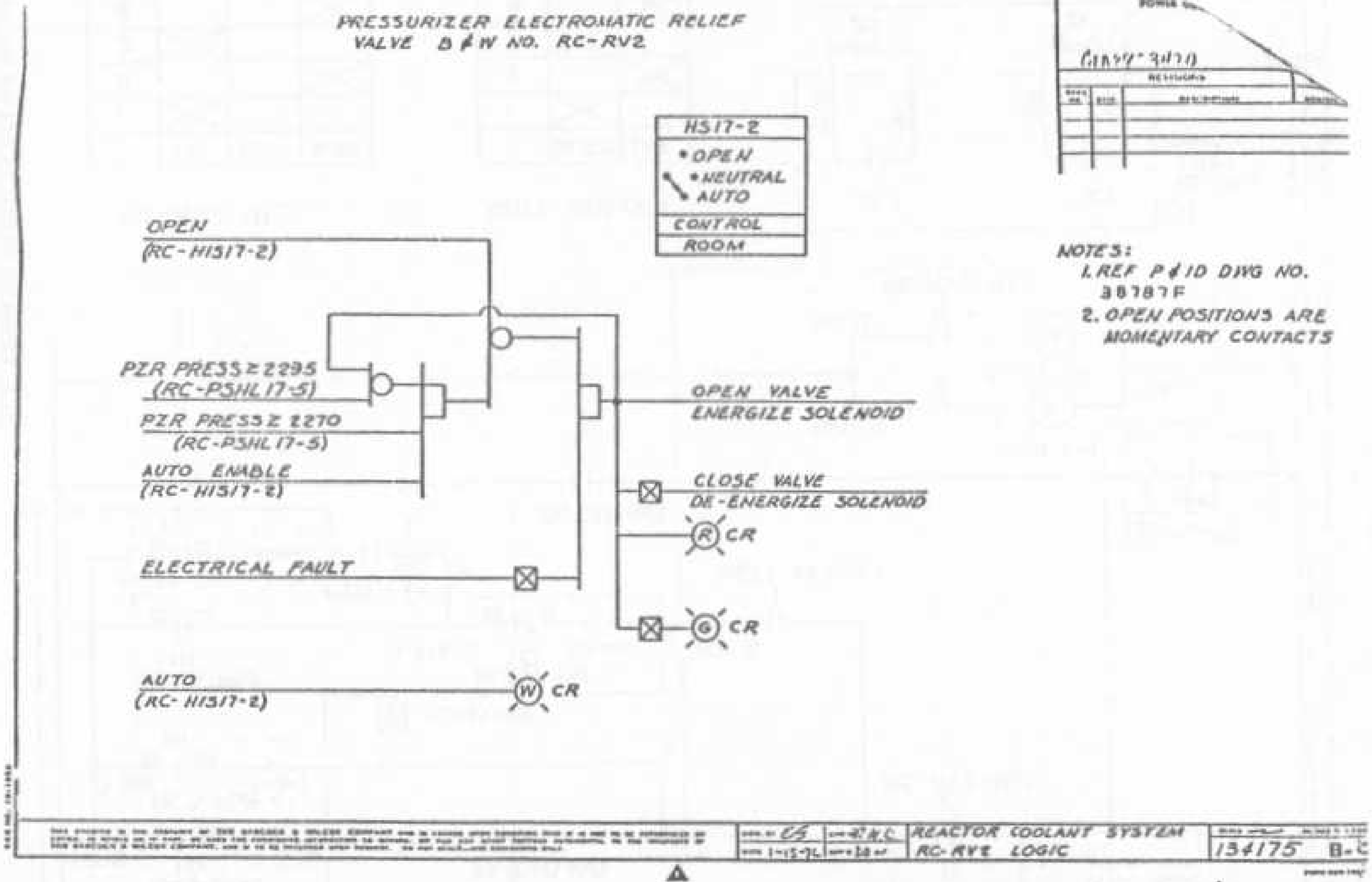
3 MILE ISLAND STATION - UNIT NO. 2

ELEMENTARY DIAGRAM

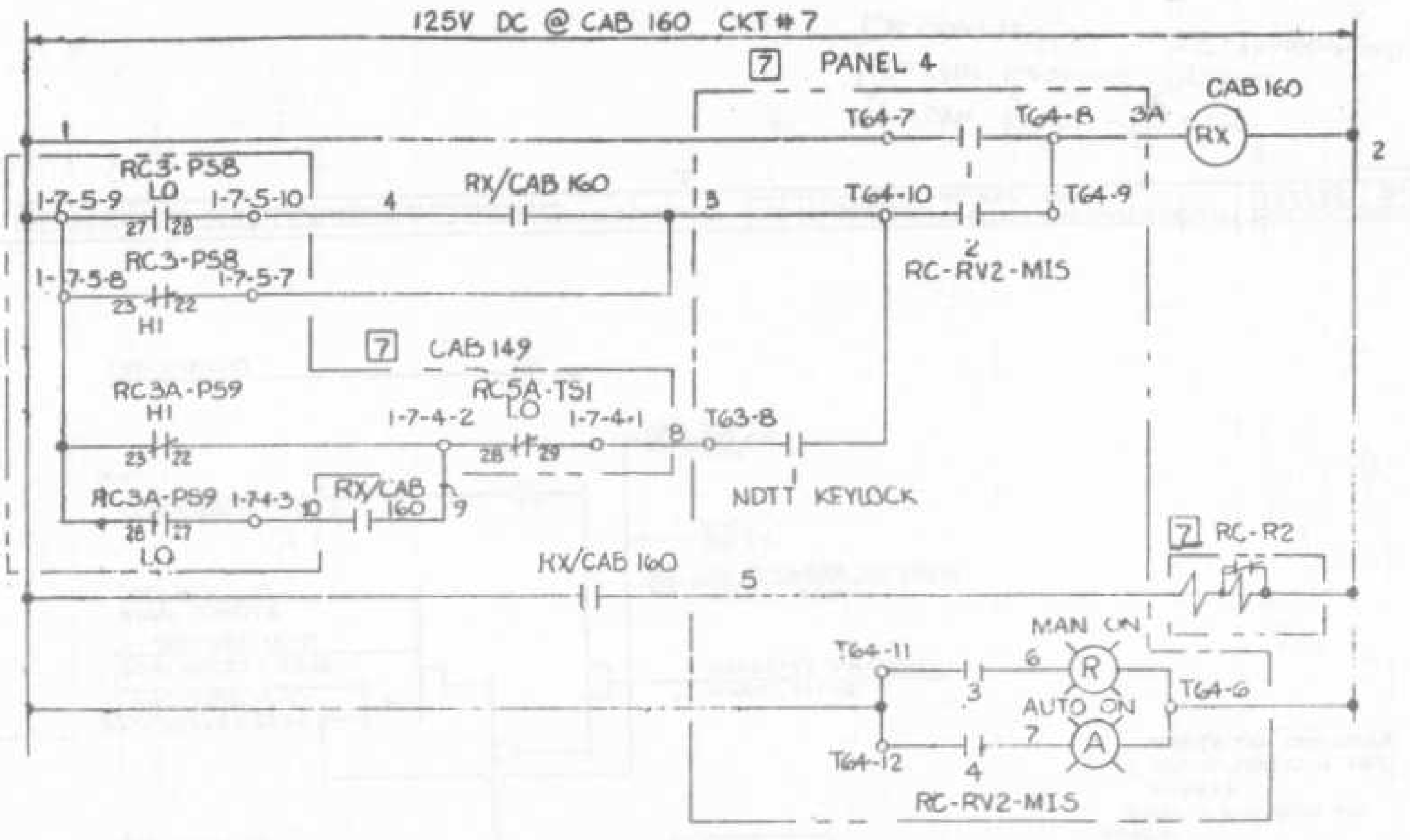
PILOT ACTUATED VALVE

113458 C 1

ATTACHMENT 13



From B&W Standard
Reactor Coolant System
Description - 15-1000000-06

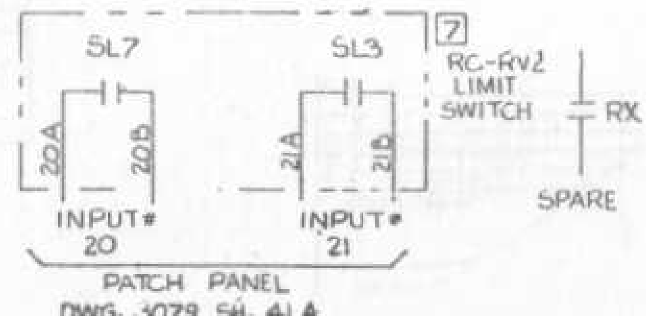


RC-RV2-MIS

	ON	OFF	AUTO
1	X		
2			X
3	X		
4			X

NDTT KEYLOCK

	AUTO	OFF
1	X	
2		X
3	X	
4		X



ELECTROMATIC PRESSURE RELIEF VALVE
THREE MILE ISLAND NUCLEAR STATION UNIT 2

ATTACHMENT 14

5

TMI UNIT NO. 2
BURNS AND ROE, INC.
ENGINEERING CHANGE MEMO
(REV 12/75)

REV.1 N/A
COST EST. $250.-
B&R DATE 4/6/78
GPU DATE 4/6/78

SERIAL NO. S-5934 REV1
DISCIPLINE ELECTRICAL
SUBJECT RC-RV2 "ON" SIGNAL LIGHT
ATTACHMENTS 3

REFERENCES:
PR 2718 REV1. PR 2731
3079-14 R12
3301-22 R14
7-09-002 (B&W E3045977K1)
3339 IC-5 R3
3339-3 R5
3333-1 DR3

AREA:
+ FIELD SOLUTION

DISTRIBUTION
UE&C
JOB ENGINEER(2)
GEN SUPT.
SUPT. UNIT 2
QC SPVSR.
GPU
ASST. PROJ. MGR.
CONST. ENG.(2)
QA SPVSR.
B&R
SEE PP-74

CHANGE:
REVISE WIRING ON PANEL 149 AS NOTED ON ATT # 1, 2 & 4
ADD RED INDICATING LIGHT (GE-ET16), NAME PLATE AND ASSOCIATED WIRING AS NOTED ON ATT. # 5, 6 & 7
REVISE TERMINATION OF CABLE M920C AS NOTED ON ATTACHED REVISED TERM. SLIP
REV.1 DO NOT CHANGE FAIL MODE OF HI-LO MONITOR 4-4-12. (RC3A-PS9)

+
DESIGNER APPROVAL DATE

REASON FOR CHANGE:
TO PROVIDE "OPEN SIGNAL ON" INDICATION FOR VALVE RC-RV2 AND TO REVISE LOGIC TO HAVE RC-RV2 TO FAIL CLOSE ON LOSS OF POWER.
REV.1 FAIL MODE CHANGE FOR RC3A-PS9 IS NOT REQUIRED FOR PROPER OPERATION OF SYSTEM.

[signature] 4/6/78 B&R ENGINEER DATE
R. Brownwell PROJECT ENGINEER
REV.1 [signature] 4/14/78 R. Brownwell

FOLLOW-UP ACTION
REVISE SPEC. NO
REVISE DWG. YES

ACKNOWLEDGEMENT:
UE&C - NAME
DATE

DISCIPLINE SUPT RESPONSIBLE FOR IMPLEMENTATION 241
NAME DISCIPLINE
R.J. TOOLE GPU TEST SU

~~BR~~ ECM NO. 5-5934 REV.1
ATT. 1 OF 23.
REF. DWG. —

PROPOSED RESOLUTION

B&W has reviewed the electromatic relief valve logic and agrees to the concept of having relief valve fail closed on loss of NNI power supply to the Hi-Low Monitor (3-10-12). To achieve this condition, switch S-1 should be in the deenergized mode and the wiring modification be made as indicated in the attached sketch. Per your request, a formal field change will follow.

To provide an indication that the electromatic relief valve has an open signal, a review of the construction schematics indicates that a control room indicating light operated from power to the solenoid can be added without additional cabling. (Refer to B&R drawing #3079, sheet 14.) This light could be actuated by the same auxiliary relay in the power distribution panel that supplies power to the valve solenoid.

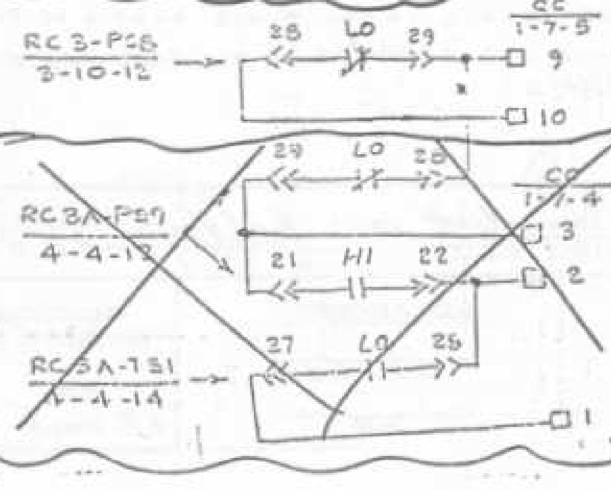
~~B&W CONSENT & FC REQ'D FOR CHANGES~~

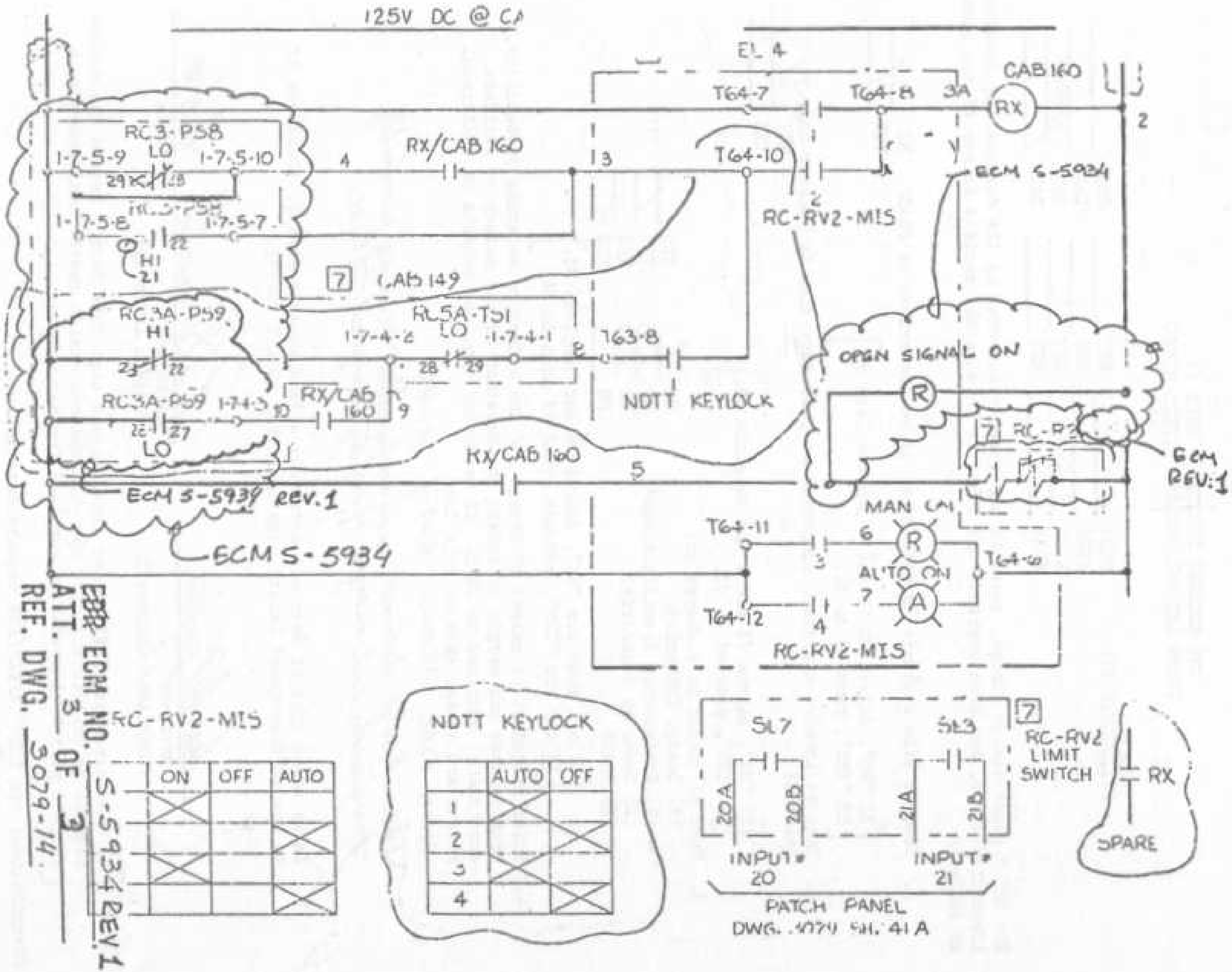
B&R COMMENT

THE CHANGES PROPOSED BY B&W ARE INCOMPLETE BECAUSE LOSS OF POWER TO RC3A-PS9 & RC5A-TS1 CAN ALSO INITIATE ELECTROMATIC RELIEF VALVE TO OPEN. HENCE FOLLOWING ADDITIONAL CHANGES ARE REQUIRED TO MODIFY RC3A-PS9 & RC5A-TS1 USED TO CONTROL NDTT BEHAVIOUR.

1. CHANGE SWITCH S1 TO DEENERGIZED POSITION - IN MODULE 4-4-12 (RC3A-PS9) & 4-4-14 (RC5A-TS1).
2. REWIRE MODULES 4-4-12 & 4-4-14 AS SHOWN BELOW:

REV. 1.




125V DC @ CA
EL 4
CAB 160
RX
T64-7
T64-8
3A
2
RC3-PS8
LO
1-7-5-9
1-7-5-10
29K
RX/CAB 160
4
3
T64-10
ECM S-5934
1-7-5-8
1-7-5-7
HI
21
22
RC-RV2-MIS
7
CAB 149
RC3A-P59
RC5A-TSI
1-7-4-2
1-7-4-1
28
29
T63-8
OPEN SIGNAL ON
R
NDTT KEYLOCK
RC3A-P59
1-7-4-3
10
RX/CAB 160
9
27
LO
RC-R3
ECM REV:1
5
ECM S-5934 REV. 1
ECM S-5934
MAN ON
T64-11
6
T64-6
AUTO ON
A
T64-12
RC-RV2-MIS
EBR ECM NO. S-5934 REV 1
ATT. 3 OF 3
REF. DWG. 3079-14
RC-RV2-MIS
ON
OFF
AUTO
NDTT KEYLOCK
AUTO
OFF
SL7
SL3
20A
20B
21A
21B
INPUT# 20
INPUT# 21
PATCH PANEL
DWG. 3079 SH. 41A
RC-RV2 LIMIT SWITCH
RX
SPARE

TMI NUCLEAR STATION - UNIT #2
BURNS AND ROE NUCLEAR SAFETY REVIEW

ATTACHMENT TO:			
	PCN#	REV	DATE
	FCR#	REV	DATE
	ECM# S-5934	REV 1	DATE 4/14/78
	EDR#	REV	DATE

The TMI-2 plant design change, tests or experiments directed by the engineering document identified above, has been reviewed and found to involve the nuclear safety of the plant as follows:

1. Does it make a change to conditions as now described in FSAR? NO ____ YES ✓

   [If "NO", delete #2, 3 and 4 and complete #5
   If "YES", proceed to #2.]

2. Does it involve a system, component or procedure that is Nuclear Safety Related? NO ✓ YES ____

   [If "NO", delete #3 and 4 and complete #5.
   If "YES", proceed to #3.]

3. Does it result in a change to:

   (A) Design Criteria? NO ✓; YES ____
   (B) Engineering Specifications? NO ✓; YES ____
   (C) Codes or Standards? NO ✓; YES ____
   (D) FSAR? NO ____; YES ✓

   [If all four are "NO", delete #4 and complete #5.
   If any are "YES", provide brief of each in #4.]

4. Brief statement of extent of change and its effect on existing probability, consequences and margin of safety (or creation of new safety or accident conditions) compared to FSAR. If necessary continue remarks on additional sheets and attach them to this form.

   REVISE FSAR FIGURES 7.6-7 & 7A-44
   This change does not adversely affect existing probability consequences and margin of safety compared to FSAR

5. It is found that it does not adversely affect nuclear safety, therefore it is not an "Unreviewed Safety Question" (per 10 CFR 50.59).

[signature] Reviewed (Cognizant Engineer)

R. Brownwell Approved (Project Engineer)

4/14/78 Date

[If the statement of #5 is not true, do not sign this form. Return the entire package to Project Engineer who will convene design review conference with project personnel needed to complete further action.]

Distribution: Per PCN/FCR/ECM/EDR requirements and
R.A. Klingaman, Met-Ed, Reading, PA

SITE PROBLEM REPORT BABCOCK & WILCOX

CUSTOMER	ORIGINATOR / DATE	DOC. ID. CONT. NO.	SPR NO.	REV. NO.
Jersey Central	L. Rogers 4/20/78	13 - 620-0006	183	0

VENDOR	P.A. NO.	PART NO./TASK NO. GROUP NO. SEQ. NO.
Bailey Meter Company		22 -001-001

TITLE (MAX 30 CHARACTERS)	PROBLEM CONTACT
RC-RV2 Failed Open: Reactor Trip I	S. P. Maingi 4/20/78

PROBLEM IDENTIFICATION

DESCRIPTION OF PROBLEM:

SEE ATTACHED

STATUS-ACTION TO DATE, INCLUDING PERSONS CONTACTED:

SEE ATTACHED

FURTHER ACTION RECOMMENDED BY SITE PERSONNEL:

SEE ATTACHED

INFORMATION ONLY

RESOLUTION

RESOLUTION: BURNS & ROE MADE CHANGE TO CLOSE VALVE ON LOSS OF POWER. B&R ISSUED ENGR. CHANGE MEMO TO HAVE CHANGE MADE. NO FURTHER ACTION IS REQUIRED FROM B&W. (FROM BOB CUTLER - GPUSC). LRPLETKE

PREPARED BY	DATE	APPROVED BY	DATE
REVIEWED BY	DATE		

COST CATEGORY	FIELD CHANGE REQ	F.C.A. NO.	SIGNIF. DEFICIENCY
☐ NORM OTHER ☐	☐ YES NO ☐	04-	☐ YES NO ☐

COMPLETION

SITE COMPLETION REPORT:

DEVIATIONS:
☐ NONE

DATE COMPLETED:

COMPLETED BY

D1: ;C!f T"rio1 OF PROM;-.1-1

On 29 March 19'¦3 at '1 1437 hours, the Ti -111-Il reactor tripped on pumps power trip followed by *rapid dcprcszurization of the Reactor* Coolant System. The reactor coolant low pressure *trips* annunciated within 73 seconds and the emergency lIP injection started in about 2 minutes following the reactor trip.

The cause of the trip **was traced to** *deenergizin,* **of vital** *power supply* **2-1V.**

(a) **Vital Bus 2-1V feeds** the RCP-1A *monitoring* c+.rcuit. Since RCP-2A was already down for clutch repairs, the loss of power to the RCP-lA monitoring circuit registered no pumps operating **in "A"** loop and. hence the signal to trip the reactor.'

(b) Vital Bus 2-1V also supplies power to the X bus .for non-nuclear instrumentation. Because of loss of X **bus' to** URI, the electromagnetic relief valve, '.'.C-RV2, received an open coz and, which initiated a rapid system depressurization.

(c) **The electromagnetic relief valve,** RC -R'12, does not have *valve indication in* the Control Room, so. the Control Room operator was unware that RC-RV2 had opened; hence, the operator did *not* take.the remedial action of closing the electromagnetic relief valve isolation valve, RC V2.

(d) There exists an apparent anomaly **in** the logic for the operation of NaOH tank valves connected to B'''5 ` lines that feed the MU pumps *suction. Due to this logic,* NaOll was fed into the suction lines of MIU pumps during the high pressure injection, which ensued after rapid depressurization.

(e) The circumstances v ich led to t:.c deenergization of vital power supply 2-1V are enumerated in the i4et-Ed Reactor Trip' Report (copy attached for reference).

STATUS - ACTION TO DATE.

GPU/Met-Ed are sorting the related problems as follows:

(a) The reactor building isolation and cooling surveillance procedure is being revised to the effect that they do not disconnect the' alternate source of power to vital buses.

(b) The logic for operating NaOH tank valves during HP injection is being reconsidered. The Reactor Coolant System chemistry was brought back to specif'fations.

(c) The electroaatic fail open logic is being questionea.

Tom-Scott of Nuclear Service and Bob Burris of Control Analysis were informed. The apparent consensus was that the electromagnetic relief valve should not fail open but should fail closed. In the safety analysis, no credit was taken for the relieving capability of the electromagnetic valve. The code safety valves exist to take care of the pressure transients.

On request by Ron Toole, GPU Test Superintendent, a logic change was suggested to GPU after consulting Doug Kemp of Engineering. A copy of GPU Problem Report 2718 is attached for reference.

It was also suggested that RC-RV2 open-close signal status lamp be wired to operatcrconsole. Burns & Roe is working on this aspect.

FURTHER ACTION RECM^1ENDED BY SITE PRSOI:J

As requested by Ron Toole,.a formal field change is being issued to modify true fail open logic of RC-RV2,and the desirability of having the key switch in MII cabinets at location IC-5-lk for testing auto operation of RC--"2 shn:'d d be reconsidered.

ATTACHMENT 16

THE INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS, INC.

POWER ENGINEERING SOCIETY

NUCLEAR POWER ENGINEERING COMMITTEE

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(704) 373-4770

April 19, 1976

To: Members of NPEC

From: I.M. Jacobs, Chairman
SC-5 Reliability

Subject: PROBABILISTIC ANALYSIS AS AN ALTERNATIVE TO THE SINGLE FAILURE CRITERIA

Subcommittee 5 of the IEEE Nuclear Power Engineering Committee is endeavoring to assist the nuclear industry in using probabilistic studies and reliability techniques in the design and analysis of nuclear power plants, and to increase the acceptability of such studies to the Nuclear Regulatory Commission. We have found that some widely accepted criteria are often applied to designs in such a way that the reliability or availability of the affected system is not improved, and may even be decreased. The single failure criterion is one such criterion. There is no question that the concept of redundancy, implicit in the single failure criterion, is of great importance in the design of high reliability systems. However, the criterion, as sometimes stated, has no limits or qualifiers on the credibility of likelihood of failure or the required system reliability. The criterion is thus sometimes applied in such depth, and to such improbable situations, as to defeat the original purpose. One can indeed be led into a higher cost/lower reliability system by overemphasis on "single failure." In contrast, a probabilistic evaluation allows a balanced judgment of the reliability actually needed and allows the designer the option of defense of non-redundant components in cases where they may indeed be adequate.

The Subcommittee, therefore, recommends that in future standards which invoke the single failure criterion, the option of performing probabilistic assessment to demonstrate an equivalent level of reliability or availability goals should be allowed as a alternative to applying the single failure criterion. The Subcommittee believes that such an optional approach would be used infrequently but when used would lead to a thorough and balanced study of the system involved, and the safety requirements of the system. Such a study should lead to a greater level of safety in system design than is achieved by a ritualistic application of the single failure criterion

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The option might also improve situations in which the proper application of the single failure criterion is not well defined, such as systems shared between nuclear units.

The decision to use this option would place the burden of proof on the designer or analyst because there are no standards or regulations which define acceptable methods of analysis or goals. Subcommittee 5 is, however, preparing standards for such analyses, and has issued IEEE-352, a tutorial standard detailing general methods which can be used. Forthcoming standards will include a data compilation for use in this work.

Having urged you to consider the provision of an option to the single failure criterion, the Subcommittee would appreciate your comments on the matter.

:ksg

cc: SC-5 Members

ATTACHMENT 17

EXCERPTS FROM "GENERIC ITEMS LIST" ACRS

Under the 1957 amendment to the Atomic Energy Act, the Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory body to review safety studies and facility license applications referred to it, and to make reports thereon. It is also to advise the NRC with regards to proposed or existing reactor facilities and the adequacy of proposed reactor safety standards. The ACRS recognizes the evolving nuclear technology and follows issues that seem significant by listing these issues as generic. From time to time the ACRS will submit to the NRC a status report on generic issues (Report No. 7, March 21, 1979, "Status of Generic Items Relating to LWR's"). Items 1 through 52 are all generic items considered resolved as of the date of Report No. 7. Following each resolved item is a brief statement of the specific action that resulted in resolution. Items 53 through 77 are items for which resolution on a generic basis is still pending. Formal actions, such as issuance of regulations or regulatory guides, are anticipated for many of these items. With regard to the status of generic issues, as they apply to each plant, the NRC staff addresses the status of the pertinent issues in the applicable Safety Evaluation Report (SER). The ACRS identifies those that it believes relevant in its reports on individual projects.

None of the resolved or unresolved generic items on the ACRS list refer to the PORV problem. FMEA of the PORV is not a generic item.

Finding

ACRS, being the highest and statutory technical review body in matters of reactor safety, has not recognized the PORV as safety-related, or as a reason for initiating a generic item.

ATTACHMENT 18

EXCERPT FROM "ABNORMAL OCCURRENCES REPORTED TO CONGRESS"

Section 208 of the Energy Reorganization Act of 1974 requires quarterly reports be made to Congress by the NRC on abnormal occurrences. The act identifies an abnormal occurrence as an unscheduled incident or event which the NRC determines to be significant from the standpoint of public health or safety. The long list of malfunctioning PORVs did not trigger a single abnormal occurrence report to Congress by the NRC.

The Davis-Besse-1 event of Sept. 24, 1977, was recognized by the NRC as being of some significance but not as an abnormal occurrence. It was reported in NRC's Current Events/Power Reactors published in December 1977 under the heading "Valve Malfunctions":

> On September 24, 1977, DB-1 experienced a depressurization when a pressurizer PORV failed in the open position. The RCS pressure was reduced from 2255 psig to 875 psig in approximately 21 minutes. At the beginning of this event, steam was being bypassed to the condenser and the reactor thermal power was at 263 MW, or 9.5%. Electricity was not being generated.... At approximately 21 minutes into the transient, the operators discovered that the PORV was stuck open. Blowdown via this valve was stopped by closing the block valve....
>
> The transient was analyzed by the NSSS vendor and determined to be within the design parameters analyzed for a rapid depressurization. With exception of the above noted malfunctions, the plant functioned as designed, and there was no threat to the health and safety of the general public.

NUREG-0090, Report to Congress on Abnormal Occurrences for October to December 1978, lists "Abnormal Occurrence Criteria" (Appendix A, page 12):

> The following criteria for this report's abnormal occurrences determinations were set forth in an NRC policy statement published in the Federal Register (42 FR 10950) on February 24, 1977.
>
> "Events involving a major reduction in the degree of protection of the public health and safety. Such an event would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to: . . .major deficiencies in design, construction, use of, or management controls for licensed facilities or materials.
>
> Examples of the types of events that are evaluated in detail using these criteria are:

For Commercial Nuclear Power Plants:

1. Exceeding a safety limit of license Technical Specifications (10 CFR Part 50.36 (c)).

2. Major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.

3. Loss of plant capability to perform essential safety functions...."

ATTACHMENT 19

EXCERPTS FROM "UNRESOLVED SAFETY ISSUES REPORTED TO CONGRESS"

In December 1977, the Energy Reorganization Act of 1974 was amended to include a new section 210 which requires that the NRC develop and submit to the Congress a plan for the specification and analysis of "Unresolved Safety Issues" relating to nuclear reactors. The incident at Davis-Besse may have had something to do with this amendment. In accordance with the new section 210, a report, NUREG-0410, dated December 1977, entitled "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," was submitted to the Congress, identifying 133 generic tasks and describing the NRC program that was then already in place. It is the NRC's view that the intent of Section 210 is to assure that plans are developed and implemented on issues with potentially significant public safety implications. Malfunctioning PORVs are not found among the issues considered.

NUREG-0510, January 1979, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," describes the review by NRC undertaken over the last year that resulted in identifying 17 issues as "unresolved safety issues." In addition, the report provides specific discussions of why certain issues were not included. The report also provides a brief background discussion of the NRC program for the resolution of generic issues.

Major elements of the NRC program are described in NUREG-0410 and are summarized in NUREG-0510. A Technical Activities Steering Committee was established, chaired by the deputy director of the Office of Nuclear Reactor Regulation (NRR); it includes, as members, the four division directors in NRR. The committee's functions include assigning proposed generic tasks to priority categories, assigning lead responsibility to an NRR division, approving task action-plans, and regularly reviewing progress. The Steering Committee's judgmental decisions regarding priorities is based on the interpretation of definitions setting four categories of priorities for generic technical tasks (NUREG-0510, Table 1):

Category A. (40 tasks)
Those generic technical activities judged by the staff to warrant priority attention in terms of manpower and/or funds to attain early resolution. These matters include those the resolution of which could (1) provide a significant increase in assurance of the health and safety of the public, or (2) have a significant impact upon the reactor licensing process.

Category B. (73 tasks)
Those generic technical activities judged by the staff to be important in assuring the continued health and safety of the public but for which early resolution is not required or for which the staff perceives a lesser safety, safeguards or environmental significance than Category A matters.

Category C. (17 tasks)
Those generic technical activities judged by the staff to have little direct or immediate safety, safeguards or environmental significance but which could lead to improved staff understanding of particular technical issues of refinements in the licensing process.

Category D. (3 tasks)
Those proposed generic technical activities judged by the staff not to warrant the expenditure of manpower or funds because little or no importance to the safety, safeguards or environmental aspects of nuclear reactors or to improving the licensing process can be attributed to the activity.

REFERENCES

1. NUREG-0600, "Investigation into the March 28, 1979, Three Mile Island Accident by the Office of Inspection and Enforcement, NRC," August 1979.

2. Ronald Eytchison, "Technical Assessment of Operating, Abnormal, and Emergency Procedures," staff report of the President's Commission on the Accident at Three Mile Island, October 1979.

3. Jasper L. Tew, "Summary, Sequence of Events," staff report of the President's Commission on the Accident at Three Mile Island, October 1979.

4. NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company (Tedesco Report)," May 1979.

5. ORNL-TM-3782, "The Selection and Procurement of Relief Valves for LWR Systems," (7140047), June 1972.

6. B&W System Requirement Specifications, June 1976 (BWNP-20004), for Reactor Coolant Systems (829008).

7. "Application of the QA Criteria (Appendices A/B issue)," NRC memorandum, Morrison to Haass, July 24, 1979.

8. William Bland and Dwight Reilly, "Quality Assurance," staff report of the President's Commission on the Accident at Three Mile Island, October 1979.

9. NSMB-537, Attachment for 6.1 (included in Appendix A)

10. Communication from Valve Expert, Will Bush, ORNL (615-574-6367).

11. NUREG-0462, "Technical Report on Operating Experience with BWR Pressure Relief Systems (8140030)," July 1978.

12. "Post TMI-2 Review of Westinghouse and Combustion-Engineering Operating Plants," task force report by Ashok C. Thadani to D. Ross, Jr., July 6, 1979.

13. LER output on events involving safety/relief valves from 1969 to the present, (7140001), July 21, 1979.

14. NRC, Current Events/Power Reactors, March 1 to April 30, 1978.

15. NRC memorandum, Eisenhut to Stello, March 30, 1978.

16. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations (Mattson Report)," July 1979.

17. John MacMillan testimony at House of Representatives hearings May 24, 1979.

18. NUREG-0161, "Instructions for Preparation of Data Entry Sheets for LER File (8170002)," July 1977.

19. EMD-79-16, Report by the GAO, "Reporting Unscheduled Events at Commercial Nuclear Facilities," Jan. 26, 1979.

20. NUREG/NSIC-171, "Operating Experiences with Valves in LWR Nuclear Power Plants for Period 1965-1978 (8290011)", July 1979.

21. NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," Report to Congress, January 1979.

22. Milt Lewison, quoted in Nucleonics Week, June 14, 1979.

23. NUREG-0321, Sandia's review of NRC's Quality Assurance Program, August 1977.

24. ACRS to NRC "Report No. 3 on TMI-2," May 16, 1979.

25. ACRS to NRC, "Report No. 7, Status of Generic Items Relating to LWRs," March 21, 1979.

26. NUREG-0090, "Report to Congress on Abnormal Occurrences for October to December 1978."

27. NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," December 1977.

28. NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," January 1979.

APPENDICES*

A. NSMB-537, agenda and attachments for ANSI's NSMB meeting, Sept. 14, 1979. Accession #1013019.

B. ANSI B16.41, Proposed Standard, May 1979, "Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants." Accession #1013020.

C. Met Ed Audit Report 77-33, "Vendor Audit: Dresser Industrial Valve and Instrument Division," Oct. 10, 1977. Accession #1013021.

D. Sava Sherr/Mel Green Agreement Between IEEE and ASME, June 1975, as amended. Accession #1013022.

E. Regulatory Guide, Draft of June 1979, "Recommendations for Inservice Testing of ESF, Containment, and Safety Valves in LWRs." Accession #1013023.

F. B&W memorandum, W. Spangler to G. Olds, May 21, 1979, "History of PORV Malfunction." Accession #1013024.

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I. F. Forscher, Trip Report to Nuclear Reliability Committee Meeting, Aug. 6-7, 1979, Phoenix, Ariz. Accession #1013027.

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*These documents are part of the Commission's permanent records that will be available in the National Archives.

REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

CONTAINMENT
TRANSPORT OF RADIOACTIVITY
FROM THE TMI-2 CORE TO THE ENVIRONS

BY

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I. EXECUTIVE SUMMARY

The major radioactive releases from the Three Mile Island Unit 2 (TMI-2) accident were airborne noble gas fission products -- xenon and krypton -- as well as a small fraction of the radioactive iodine isotopes. These isotopes, in addition to other fission products, were dissolved in the reactor primary coolant water. It is believed that the major pathway of radioactivity release from the primary system to the auxiliary building was through the reactor coolant let-down/make-up system.

The pressurization of the reactor coolant drain tank by the blow-down from the pilot-operated relief valve (PORV) pushed water from the drain tank through the reactor building vent header into the auxiliary building vent header. It is believed that there was sufficient pressure imposed on the auxiliary building vent header to damage some component, probably the water traps incorporated for drainage of the system. The damaged components were future leakage points.

The dissolved gases were released from coolant during depressurization of the let-down fluid. The relief valves just downstream of the block orifice valve (pressure reduction of 2,135 pounds per square inch guage (psig) to about 20 psig) and of the make-up tank discharged into the reactor coolant bleed holdup tanks. This in turn pressurized the reactor coolant bleed holdup tanks and the pressure relief valve on these tanks lifted, venting the gases to the relief valve vent header. The relief valve vent header has a direct and unencumbered pathway to the station vent. The make-up tank also was vented to the vent header in the auxiliary building. The radiation monitoring systems showed a direct correlation of these ventings with releases, probably through damaged components in the vent header system. The ventilation systems of both the auxiliary building and the fuel handling building transported the released gases out the station vent.

The following observations should be noted:

- The discharges of pressure relief systems that communicate with the primary coolant were not routed to the reactor containment system. Examples are the discharge of the relief valves of the reactor coolant bleed holdup tanks and of the waste gas compressors.

- The reactor containment building is not isolated on radiation signals. This isolation probably would not have precluded the airborne radioactive releases at TMI-2.

- It appears that inadequate attention was given to design to assure matching component capabilities. For example, it is believed that the water traps WDG-U8A and WDG-U9A had lower pressure capabilities than the pressure relief valve WDG-R-3 of the vent header system.

- The concrete in the auxiliary building and the fuel handling building was not sealed prior to startup.

- **Readily accessible up-to-date, readable drawings and specifications were not available on-site.**

II. PURPOSE AND SUMMARY

PURPOSE

The purpose of this report was to prepare an evaluation as to how and why the radioactivity of the TMI-2 accident got from the reactor to the auxiliary building and fuel handling building and out of the station.

SUMMARY

The report presented here is an evaluation of how the radioactivity (a) left the reactor core, (b) was transmitted to the auxiliary building, and (c) was exhausted to the atmosphere.

The reactor accident basically was caused by not providing adequate cooling to the reactor core. The fuel cladding was extensively damaged and released all of the gaseous fission products, from the clad to fuel gap, to the primary coolant system. A part of those gaseous fission products are now in the reactor containment building, and a fraction of those gases escaped to the auxiliary building. Those gases that were transported to the auxiliary building were either released to the atmosphere or routed to the waste decay tank. Part of the gases in the waste decay tanks were transferred back to the reactor containment building.

There are eight pathways for the releases of airborne fission products discussed. Other pathways were reviewed and covered in referenced documents. These other pathways were not included here because they were not worthy of detailed study and were considered less significant. Of the eight considered here, only the following five had significant potential:

1. Reactor coolant let-down/make-up system -- This was the major pathway and the source of gas pressure buildup in the auxiliary building.

2. Reactor coolant drain tank vent to the vent header in the auxiliary building -- This pathway was believed to be the cause of damage to the vent header, thereby setting up significant leak paths for gases vented to the vent header.

The remaining three are considered substantially less significant:

3. Reactor coolant drain tank to the reactor coolant bleed tanks in the auxiliary building -- This added water to the inventory in the auxiliary building.

4. Reactor coolant drain tank vent to the reactor coolant drain tanks -- After the rupture disk blew in the drain tank, there was little pressure differential to push activity from the reactor containment building.

5. Reactor building sump to the auxiliary building sump -- This pathway contributed to excess water in the auxiliary building but had very little radioactivity because the water was released prior to significant core damage.

Simplified schematics of the above pathways were prepared and included in this report. To give a better understanding, drawings showing the interconnections of liquid and gas systems and the overall pathways systems were prepared and included.

A description of the radiation monitoring in the plant is provided. The evaluation of the response of the radiation monitoring showed that a very small release occurred early in the accident. After approximately 2 hours to 30 minutes after the core was partially uncovered, larger releases of radioactivity were evidenced. These releases continued for some time with the majority of the activity released in the first several days.

The most probable release pathway was through the reactor coolant let-down/make-up system. These releases were effected primarily by degassing of the primary coolant water in the let-down/make-up system and either going directly to the station vent from the pressure relief valves of the reactor coolant bleed holdup tanks, or by venting gases from the make-up tank to the vent header in the auxiliary building. The reactor coolant bleed holdup tanks were the recipient of pressure relief valves from the block orifice valve outlet and from the make-up tank.

Based on the evaluation developed in this report, the following observation is made: The discharges of pressure relief systems that communicate with the primary coolant were not routed to the reactor containment system. Examples are the discharge of the relief valves of the reactor coolant bleed holdup tanks and the waste gas compressors.

III. DESCRIPTION OF ACCIDENT

The Nuclear Regulatory Commission (NRC) (NUREG-0600), Metropolitan Edison (Met Ed) General Public Utilities (GPU) (Preliminary Annotated Sequence of Events, March 28, 1979; July 16, 1979, Rev. 1), and the Nuclear Safety Analysis Center (NSAC) of the Electric Power Research Institute (EPRI) (NSAC-1), have published extensive descriptions of the accident (reference 1, 2). The discussion here is to highlight events that might have consequences related to the release of radioactivity.

The plant was operating at 97 percent power at 4:00:37 a.m. on March 28, 1979 (reference 3). Reactor primary coolant system pressure was 2,155 psig. Reactor coolant make-up pump B (MU-P-1B) was in service supplying make-up and reactor coolant pump seal injection flow. Normal reactor coolant system let-down flow was approximately 70 gallons per minute (gpm) (reference 4). Flow recorder FR 7100 indicated that every 2 to 3 hours, liquid was being pumped from the reactor coolant drain tank to the reactor coolant bleed holdup tanks, indicating that the cooling system was also in operation (reference 5). The reactor primary coolant had about 0.397 HO1/cc radioactivity. There were problems in transferring resins in the standby demineralizer of the condensate polishing system. The fuel handling building supply exhaust fans were in service. The auxiliary building exhaust fans were in service. The status of the auxiliary building supply fans is not known.

The following is an abbreviated sequence of events, as reported in reference documents.

Item (Time after Feed Water Pump Trips)	Event
0 (4:00:37)	Feed water pumps trip; main turbine and main generator trip.
3 sec	Electromatic relief valve opens (2,255 psig).
4 sec	Pressure started to increase in reactor coolant drain tank.
8 sec	Reactor trips on high pressure (2,346 psig).
12 sec	Signal that electromatic valve should have closed.

Item (Time after Feed Water Pump Trips)	Event
13 sec	Make-up pump IA (MU-P-lA) was started, and a high pressure valve was opened. (Make-up pump lB was still operating.)
30 sec	The reactor coolant low-pressure trip setpoint was reached (reactor pressure at 1,940 psig).
39 sec	Make-up pump lA tripped.
41 sec	Make-up pump 1A was restarted.
60 sec	Reactor coolant drain tank pressure at 12 psig and increasing.
1 min, 26 sec	Reactor coolant drain tank temperature 85.5!F and increasing.
2 min, 1 sec	High pressure injection system automatically started at reactor primary coolant pressure nominal setpoint of 1,600 psig. Make-up pump 1B trips automatically, and make-up pump 1C starts. Make-up pump IA is still running.
3 min, 13 sec	Reactor coolant drain tank relief valve lifted at 120-122.
3 min, 13 sec	Manual bypass of high-pressure injection system controls.
3 min, 26 sec	Reactor coolant drain tank high temperature alarm. (As stated in the introduction, the cooling system was in operation, but evidently could not keep up with the heat input from the escaping water of the primary system through the electromatic relief valve.)
4 min, 38 sec	High pressure injection throttled.
4 min, 58 sec	Let-down flow increased to rate greater than 160 gallons per minute (reference 6). (This is accomplished by remotely opening a bypass valve, MU-U5, around the block orifice valve MU-1-E in the reactor coolant let-down/make-up system.)
6 min, 54 sec	Let-down cooler 1A outlet temperature alarms high at 139°F.
6 min, 58 sec	Reduced let-down flow to 71.4 gallons per minute by closing.

Item	Event
(Time after Feed Water Pump Trips)	
7 min	Reactor containment building purge air radiation gas monitors HP-R-225 and HP-R-226 indicate small increases in radioactivity (reference 7).
7 min, 29 sec	Reactor building sump pump A (WDL-P-2A) starts.
8 min, 18 sec	Emergency feedwater block valves were opened.
10 min	Reactor building sump pump B (WDL-P-2B) started.
10 min, 48 sec	Reactor building sump high-level alarm (4.65 ft).
14 min, 48 sec	Reactor coolant drain tank rupture disc WDL-U26 failed at 192 psig (reference 8). (This is an 18-inch vent line.)
15 min	Radiation monitor on hydrogen purge HP-R-229 alarmed on iodine channel (reference 9).
16 min	Reactor building air sample radiation monitor HP-R-227 alarmed on gas, particulate and iodine channels (reference 10).
19 min	Station vent monitor HP-R-229 alarmed on gas channel (reference 9).
22 min	Radiation monitors HP-R-221A (particulate), HP-R-225 (gas), HP-R-226 (gas), HPR-225 (gas), HP-R-226 (gas), HP-227 (gas), and IC-R-1092 showed increased radiation levels (references 7, 9, 10, 11).
28 min	Radiation monitor HPR-225 (gas) (reference 7). The evaluation by NRC and Med Ed included responses by several other monitors showing the same pattern of increases.
38 min	Reactor building sump pumps A and B were stopped.
60 min	Let-down cooler A radiation monitor 1C-R-1092 increased (reference 11).
1 hr, 13 min	Reactor coolant pumps 1B and 2B were stopped.
1 hr, 23 min	Let-down cooler A radiation monitor IC-R-1092 increased again.

Item	Event
(Time after Feed Water Pump Trips)	
1 hr, 41 min	Reactor coolant pumps 1A and 1B were stopped.
1 hr, 51 min	Reactor coolant loops A and B hot leg temperatures increasing.
2 hr, 22 min	Reactor containment building air sample monitor HP-R-227 gas channel starts increasing again (reference 10).
2 hr, 22 min	Reactor coolant system pressure begins to rise (680 to 2130 psig over the next 41 minutes).
2 hr, 25 min	HP-R-227 (see Item 37) particulate channel started to increase again (reference 10).
2 hr, 31 min	Radiation monitors start to rise in the reactor containment building followed by radiation monitors rising in the auxiliary building about 10 minutes later. (See Section IX of this report for specific radiation monitors and times.) (References 7, 9, 10, 11, 12, 13.)
2 hr, 54 min	Started reactor coolant pump 2B.
2 hr, 56 min	Site emergency declared.
3 hr, 12 min	Electromatic relief block valve RC-V2 was opened and was closed at 3 hr 17 min.
3 hr, 19 min	High pressure injection was started manually.
3 hr, 23 min	General emergency declared.
3 hr, 29 min	Fuel handling building air exhaust fans flow was zero. NOTE: During the next 2 1/2 hours, the exhaust fans were turned on and off several times with run times of 30 to 60 minutes (references 14, 15).
3 hr, 30 min	Electromatic relief block valve RC-V-2 was closed.
3 hr, 51 min	Electromatic relief block valve RC-V2 was opened.
3 hr, 56 min	Reactor containment building isolated by high pressure signal (approximately 4 psig). Each isolation valve must be reset by operator action to put any system back into service which penetrates reactor containment.

From the time at which the reactor containment building was initially isolated at 7:56 a.m., it appears that the only pathways for radioactivity to leave the reactor containment building were through the let-down/make-up system, including the reactor coolant pump seals.

There were radiation releases by depressurization of the let-down water of such dissolved gases as hydrogen, krypton, and xenon. Some fractions of the radioactive iodine also became airborn from the water. The evolution of dissolved gases pressurized the let-down/make-up system, including the make-up tank and the reactor coolant bleed holdup tanks. The released gases were also transmitted to the waste gas decay tanks.

IV. DESCRIPTION OF CORE BEHAVIOR

The following discussion is primarily from the standpoint of core behavior and its relationship to the release of radioactivity, particularly fission gases and volatile fission products. This discussion is also qualitative and is based on what one would expect when a hot core became uncovered in a steam atmosphere.

Early in the accident, stress conditions of the fuel pins appeared to cause a release of a small amount of radioactivity (reference 7) that was picked up by radiation monitors in the plant (see Section IX). This initial release for all intents and purposes was insignificant from a health standpoint.

Little further damage was imposed upon the core until the lack of liquid phase cooling water became important at about 6:00 a.m. to 6:15 a.m. on March 28, 1979. At this time, with the reactor coolant pumps off and the primary coolant inventory insufficient, the core started to become uncovered.

At near 5:45 a.m., the out-of-core nuclear instrumentation (reference 16) indicated increased flux level that can be rationalized as decreased coolant in the core and downcomer annulus. The temperatures in the hot legs of both loops A and B of the reactor primary coolant system were increasing at about 5:50 a.m. and went greater than 620°F (reference 3). Up to this time, the core cladding would have normally stayed within several hundred degrees Farenheit of the water-steam mixture. However, when the liquid phase is not present to remove heat by vaporization, cladding temperatures rise to improve the lost heat-transfer condition.

When the cladding temperature of zircaloy reaches about 1,400 to 1,500°F, failure or cladding breach becomes a reality (reference 17). The breaching of the cladding releases fission gases not trapped in the fuel matrix. The fission gases would be krypton, xenons, and at these temperatures, iodine also would be a gas (reference 18).

Since decay was still significant, the core fuel would be expected to increase in temperature with the lack of sufficient cooling. That was indeed the case at TMI-2. At about 2,200°F, reaction of zircaloy with water molecules becomes rapid, forming zirconium oxide ($Zr0_2$) and hydrogen (H_2) (reference 17).

The zirconium oxide stays as a solid. However, the H_2 is distributed between the liquid still present and the gas phase. At the reactor primary system pressures and temperatures, the equilibrium solubility of H_2 is relatively high (reference 18). At 1,543 psig and 600°F, the mass concentration ratio of hydrogen between gases and water is 67.5 Hg H_2/g gas/Hg H_2/g liquid.

The estimates of chemical reaction of the zircaloy of the core with the water is 40 to 50 percent (reference 17).

The volume of hydrogen that was generated far exceeded the volume of fission gas. For all practical purposes, the hydrogen became the carrier gas for the fission product gases to escape out of the primary system water.

Both the fission gas and hydrogen have significant solubility in water. When the let-down water, which had these dissolved gases present under pressure, was cooled and depressurized through the block orifice valve, the dissolved gases (primarily hydrogen) were released and pressurized the auxiliary building portion of the reactor coolant let-down/make-up system. These increased pressures were the primary driving force for the fission gas releases. The releases were effected through lifted relief valves, damaged components or planned venting of specific tanks in the auxiliary building (reference 6).

V. PATHWAYS FROM THE REACTOR PRIMARY SYSTEM TO THE AUXILIARY BUILDING

LET-DOWN/MAKE-UP SYSTEM

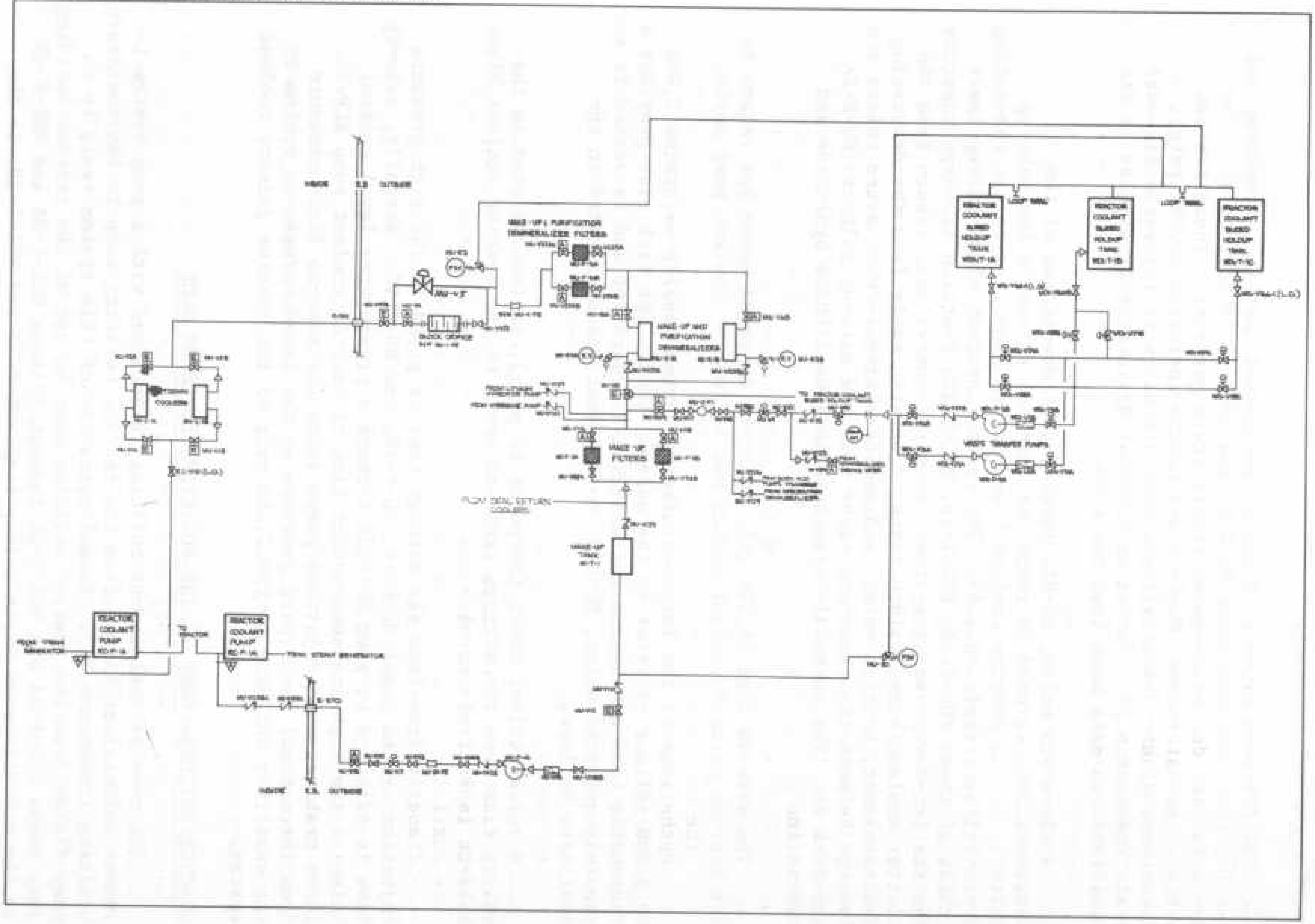
The let-down/make-up system is the major control system for water chemistry, water conditioning, and maintenance of water inventory in the reactor coolant primary system. Control of the let-down/make-up system can be either manual or automatic. During reactor operations or standby, the let-down/make-up system can be and is used to furnish conditioned make-up water for leaking seals or valves in the reactor primary coolant system (reference 6). A schematic of the let-down/make-up system is shown in Figure 1.

During normal operation, water is removed from the reactor primary cooling system from the 28-inch diameter line between the loop A steam generator RC-H-1A and reactor coolant pump RC-P1A (reference 19). The 2-1/2-inch diameter stainless steel pipe (Schedule 160) transfers the water nominally at 550° to the let-down coolers, MU-C-1A and MU-C-1B, where the primary water is cooled to approximately 120°F. The let-down coolers are located in the reactor containment building at a centerline elevation of 286 feet 3 inches (reference 20). Normal let-down flow is 45 to 70 gallons per minute with a maximum capacity of 140 gallons per minute (reference 19).

The cooled primary water at approximately 2,135 psig and 120°F exits the reactor containment building through penetration R 541 to the block orifice valve (references 6, 29). The block orifice valve is a pressure-reducing valve used to reduce let-down water from 2,135 psig to approximately 20 psig. With both reduced temperature and reduced pressure, the dissolved gas content of the primary coolant is normally sufficiently low to preclude significant degassing or two-phase flow in the let-down/make-up system. If the let-down water has significant dissolved gases, the pressure-reducing characteristics of the block orifice may not result in sufficiently low pressures in the let-down/make- up system in the auxiliary building. To prevent overpressuring filters, demineralizers, tanks and other components in the low-pressure portion of the let-down/make-up system, a pressure relief valve, MU-R-3 (setpoint -- 1130 psig) was installed just down-stream of the block orifice valve and ahead of the lower pressure components (reference 5). This pressure relief valve also protects the downstream components in the event of blockage or inadvertent closure of inline valves. With lower-than-design flows, the block orifice valve is less effective in reducing line pressure as the flow increases.

It is important to note that if the let-down/make-up system is in operation with the valves open upstream of the block orifice valve and a block or obstruction is effected downstream, this pressure relief valve MU-R3 relieves pressure by venting to the reactor coolant bleed holdup tanks (reference 6). The same relief is provided if the overpressure is caused by dissolved gases.

FIGURE 1: Let-Down/Make-up System



A remotely operated diaphragm bypass valve is installed parallel to the block orifice valve and is used to maintian adequate flows at reduced reactor coolant pressures.

The let-down water is filtered and passed through the make-up and purification demineralizers MU-K-1A and MU-K-1B (reference 6). The demineralizer can be operated singly or in parallel. There are two pressure relief valves, MU-R-5A and MU-R5B (setpoint -- 150 psig), downstream of the demineralizers that discharge to the waste disposal drain (reference 5). The waste disposal drain directs liquids to the auxiliary building sump (reference 6).

A three-way valve, MU-V8, located just downstream of the demineralizer, is used to route the let-down stream to the make-up filters, to the reactor coolant bleed holdup tanks, or to the deborating demineralizers (reference 6). The reactor coolant bleed holdup tanks (three of them: WDC-T-1A, WDK-T-1B, WDL-T-1C) function as surge capacity for the let-down/make-up system. chemical injections, return from the reactor coolant bleed holdup tanks, deborated water from the deborating demineralizer, borated water, and demineralized service water inlets are located between the three-way valve MUV8 and make-up filters, MU-F-2A and MU-F-2B. The chemical injections include lithium hydroxide and hydrazine.

The make-up tank (4,500 gal) receives filtered water for return to the reactor primary coolant system and to reactor coolant pump seals.

Operations of the let-down/make-up system usually maintains 2,800 to 3,000 gallons of water in the make-up tank. The tank also provides a reasonable time and volume for gas-liquid separation and is vented by a remotely operated valve, MU-U13, to the vent header located in the auxiliary building.

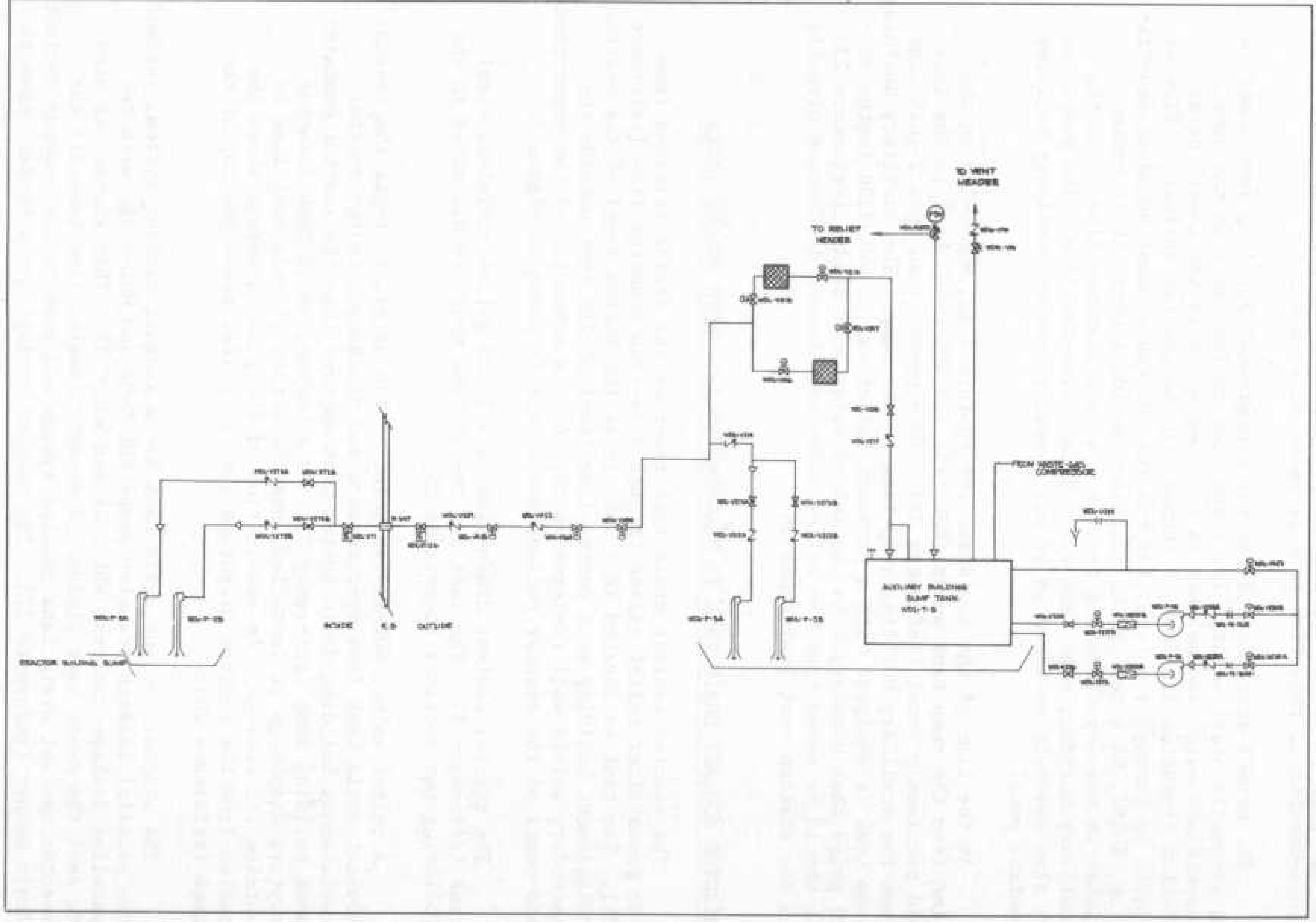
A relief valve, MU-R1 (setpoint 80 psig), is incorporated in the outlet line from the make-up tank and vents to the reactor coolant bleed hold-up tanks (references 5, 6).

The exit line from the make-up tank is piped to the high pressure injection system pumps MU-P-1A, MU-P-1B, and MU-P-1C. Normally, make-up flow is effected by pump MU-P-1B through a pressurizer level control valve to the high-pressure outlet line of reactor coolant pump RCP-1B. These high pressure injection pumps raise the make-up fluid pressure from the nominal 15-20 psig pressure of the let-down/make-up system in the auxiliary building to the 2,155 psig of the reactor primary coolant system.

<u>REACTOR BUILDING SUMP TO THE AUXILIARY BUILDING SUMP</u>

The reactor containment building is equipped with a pump system to remove uncontained liquid from the reactor building sump to the auxiliary building (reference 21). Normal operation of this system would be to pump fluids from the reactor building sump by use of the reactor building sump pumps WDL-P-2A and WDL-P-2B through filters WDL-F-8A and WDL-F-8B to the miscellaneous waste holdup tank WDL-F-3A and WDL-F-3B, to the

FIGURE 2: Reactor Building Sump to Auxiliary Building Sump System



auxiliary building sump tank. Consensus of operating personnel and level readings of the miscellaneous waste holdup tank before and after the event indicate the valve line-up was to the auxiliary sump tank. The schematic of this flow path is shown in Figure 2.

The normal operation of reactor containment building sump pumps is an automatic start on high-level and stop on low-level in the sump. Uncontained water accumulates in the sump which is the lowest point (bottom elevation 276 feet, 6 inches) in the reactor building. Flow of liquid is through a 4-inch line via reactor containment building penetration, R-547, to a tee in the auxiliary building where it is routed either to the miscellaneous tank in a 4-inch diameter line or to the auxiliary building sump tank via a 2-inch diameter line. The pumps can be also remotely switched on or off manually from an auxiliary building control panel.

At the time of the accident, the rupture disc, WDL-U-224 on the line from the sump tank pumps (WDL-P-4A and WDL-P-4B) back to the tank had previously burst (reference 22). This presents an open 2-inch line from the auxiliary building sump tank to the sump. The auxiliary building sump tank is equipped with a pressure relief valve, WDL-R200 (setpoint -- 20 psig) that discharges to the relief valve vent header (reference 23). It should be noted that the relief valve vent header discharges directly to the station vent (reference 24).

REACTOR COOLANT DRAIN TANK TO REACTOR COOLANT BLEED HOLDUP TANKS

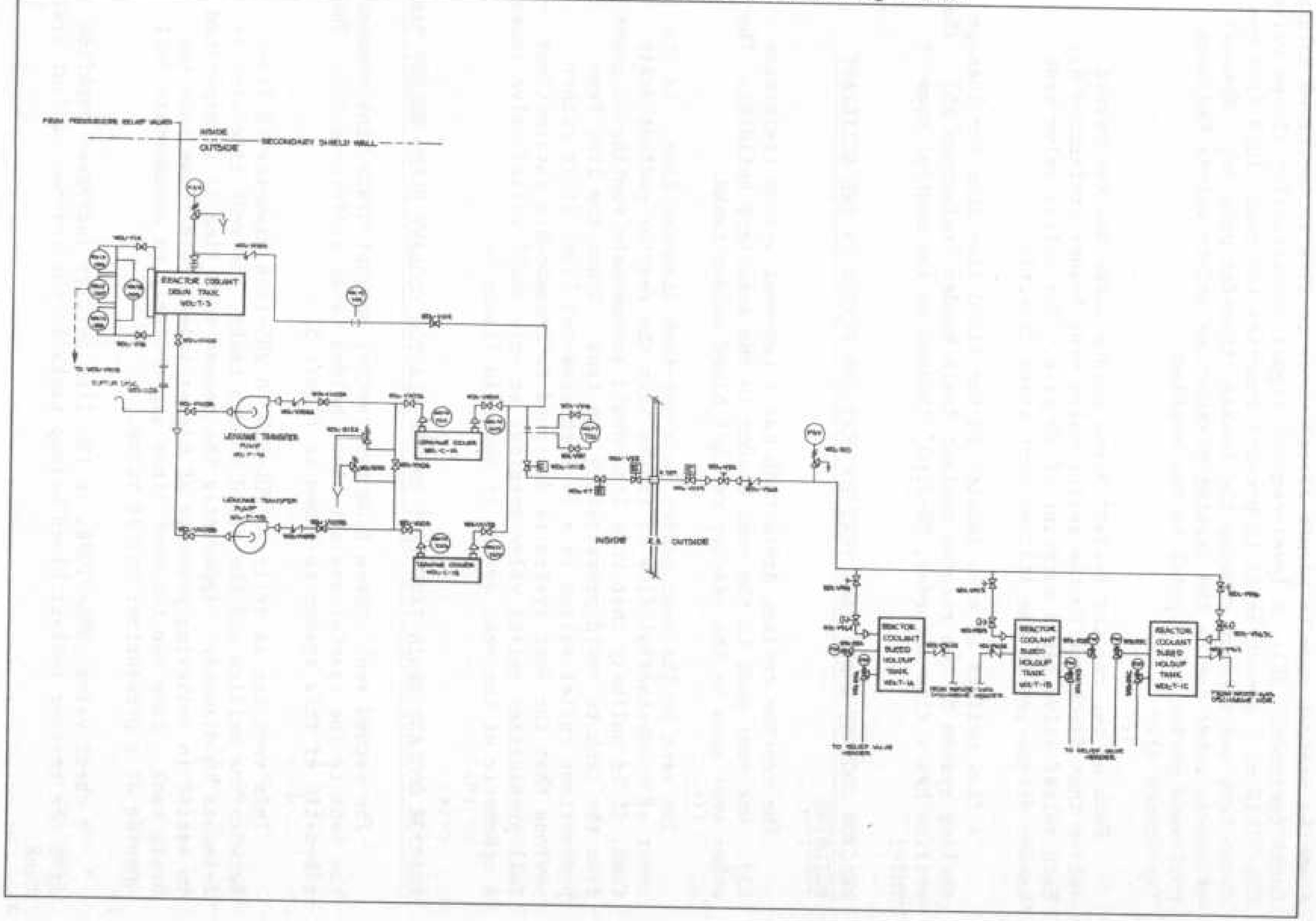
The reactor coolant drain tank receives the fluids released from the pressurizer relief valves through a 14-inch diameter line (reference 25). The tank is located in a cubicle at the bottom level of the reactor containment building at a center-line level of 289 feet outside the secondary shield wall (references 20, 26). A schematic of the connections and pumps of the reactor coolant drain tank is shown in Figure 3.

The reactor coolant drain tank is a 7,400-gallon stainless steel tank (reference 8). The tank has two systems that are discussed in the following two sections (reference 25).

A relief valve, WDL-Rl (setpoint -- 120 psig), protects the reactor coolant drain tank from overpressure and discharges to the reactor containment building drain system that empties into the reactor containment building sump (reference 5). As a backup, an 18-inch diameter rupture diaphragm is installed, and the outlet of this vent line is outside the cubicle. An examination of the plant drawings shows the outlet from the rupture diaphragm line is 7 feet above the top of the tank (reference 20).

The reactor coolant drain tank has a leakage cooling system, including two parallel leakage transfer pumps WDL-P-9A and WDL-P-9B, with two parallel leakage coolers, WDL-C-lA and WDL-C-lB. This circuit is used to cool the drain tank fluids. A 4-inch diameter line tees off the reactor coolant drain tank cooling system and goes to the reactor coolant drain header (reference 27). The reactor coolant drain header connects

FIGURE 3: Reactor Coolant Drain Tank to Reactor Coolant Bleed Holdup Tanks



to the reactor coolant bleed holdup tanks. To remove fluids from the reactor coolant drain tank, normally a valve, WDL-V1118, is open remotely from the control room to permit excess water to flow to the bleed holdup tanks (reference 28). A level-control signal automatically closes valve WDL-V1118 at a preset level to prevent removing too much fluid from the drain tank and thereby running the leakage transfer pump dry. Removal of excess water due to the leaking pressurizer safety valves had been performed periodically prior to the accident (reference 4).

Each of the reactor coolant bleed holdup tanks has two relief valves that discharge to the relief valve vent header (reference 4). Each relief valve has a setpoint of 20 psig. The relief valve vent header discharges to the station vent stack directly.

A flow orifice is also Located in the fluid line from the leakage cooling system to the reactor coolant drain header (reference 25). This orifice has a flow recorder, FR-7100, located in the control room of TMI-2.

REACTOR COOLANT DRAIN TANK VENT TO VENT GAS HEADER IN THE AUXILIARY BUILDING

The reactor coolant drain tank has a two-vent system (reference 25). One vent goes to the vent header in the auxiliary building. The other vent goes to the reactor coolant bleed holdup tanks.

The vent to the vent header is a one-inch diameter line. In the event of a substantial flow of fluids into the reactor coolant drain tank, it is unlikely that this line would accommodate venting of gases from the tank to avoid pressurizing the tank. Since the line from pressurizer relief valves is a 14-inch diameter line, it is rather obvious that the vent system is designed to accommodate intermittent full pressurizer relief valve openings or only small relief valve leaks. A schematic of the vent system is shown in Figure 4.

REACTOR COOLANT DRAIN TANK VENT TO THE REACTOR COOLANT BLEED HOLDUP TANK

The second vent system from the reactor coolant drain tank connects the tank to the reactor coolant bleed holdup tanks (reference 25). The schematic of this system is shown in Figure 5.

This vent has an orifice, WDL-U23, 0.285-inch diameter in line. Between the orifice and the bleed holdup tanks, the vent line size is 2-inches in diameter. Apparently the second vent line is incorporated to assist in relieving pressure at higher inlet fluid flows into the drain tank. Even the two vent lines are unlikely to accommodate full opening of a pressurizer relief valve.

A check valve, WDL-V1098, in the line permits increased backflow from the reactor coolant bleed holdup tanks'to the reactor coolant drain tank.

FIGURE 4: Reactor Coolant Drain Tank Vent to Vent Header In Auxiliary Building

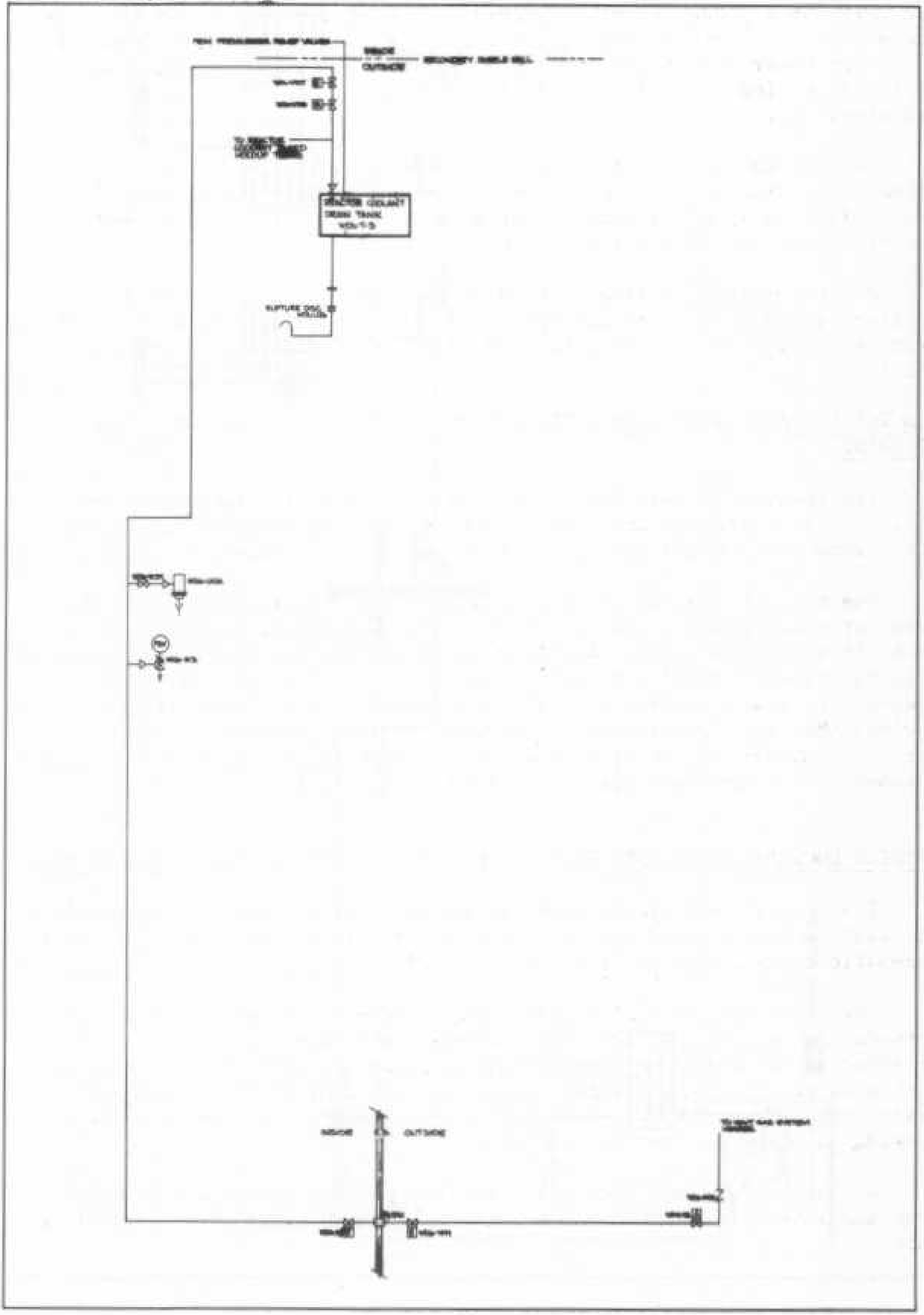
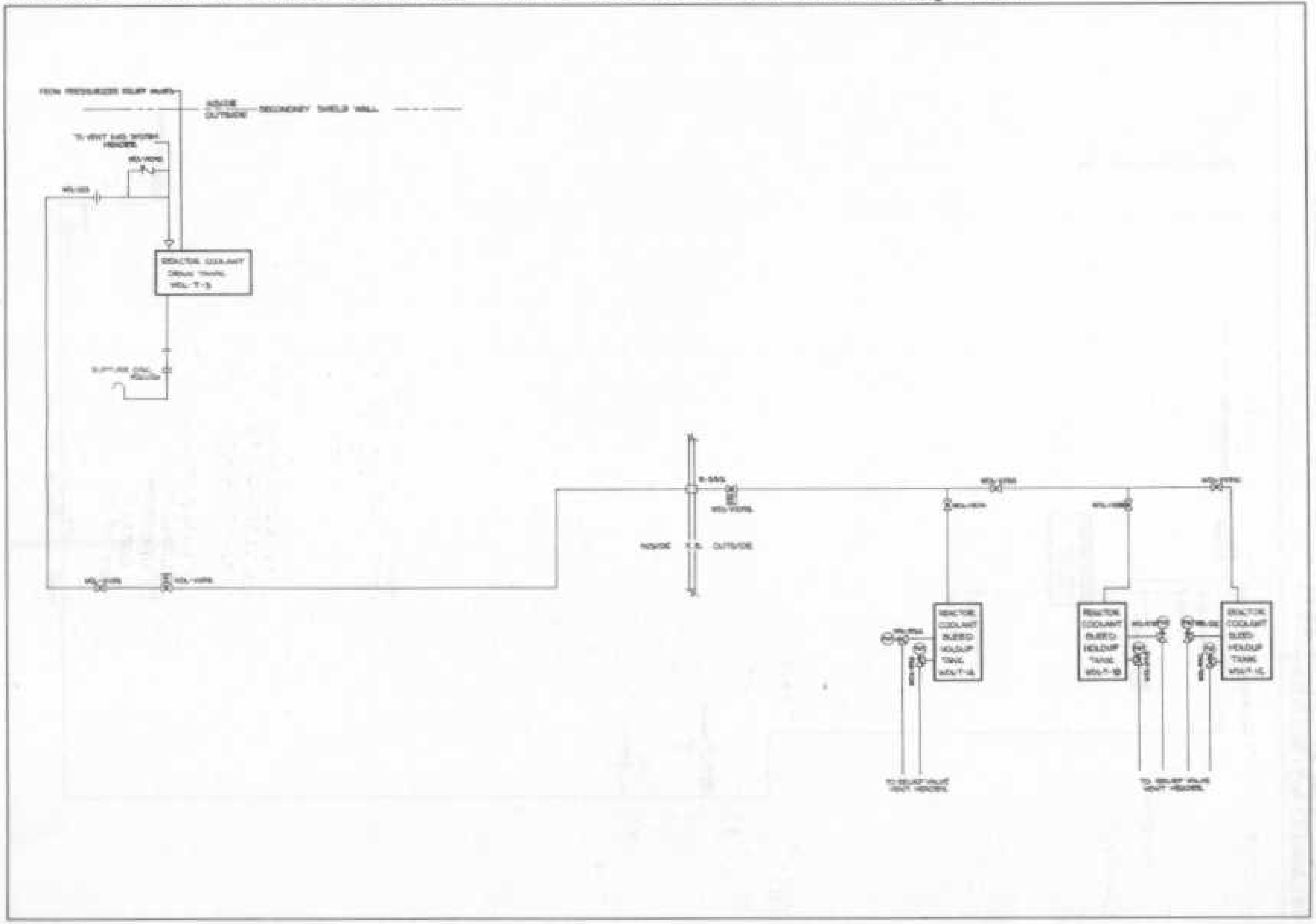


FIGURE 5: Reactor Coolant Drain Tank Vent to Reactor Coolant Bleed Holdup Tanks



REACTOR COOLANT PUMP SEALS TO SEAL RETURN COOLERS IN THE AUXILIARY BUILDING

The reactor coolant pumps have mechanical seals that are maintained at operating temperature by the primary coolant (reference 6). The internal pressure of the seals is matched by the pump seal fluid on the outside of the mechanical seal. This seal fluid is obtained from the let-down/make-up system. The return fluid is cooled to maintain low temperatures on the shaft and external seal of the pump that, in turn, extends the running time of the pump seals. Also, if there is leakage through the mechanical seal, the primary coolant leakage is caught by the seal fluid, which minimizes contamination to the reactor containment building.

Reactor coolant pump seal fluids are manifolded and exit from the reactor containment building to the seal return coolers MU-C-2A and MU-C-2B. The system is protected by a relief valve, MUR2 (setpoint -- 150 psig) that discharges to the inlet line to the make-up tank.

The seal fluid return comes from the mixture in the make-up tank. Total seal flow to each reactor coolant pump is 8 to 10 gallons per minute. A schematic of reactor coolant pump seal flow systems is shown in Figure 6.

LET-DOWN COOLERS COOLING WATER

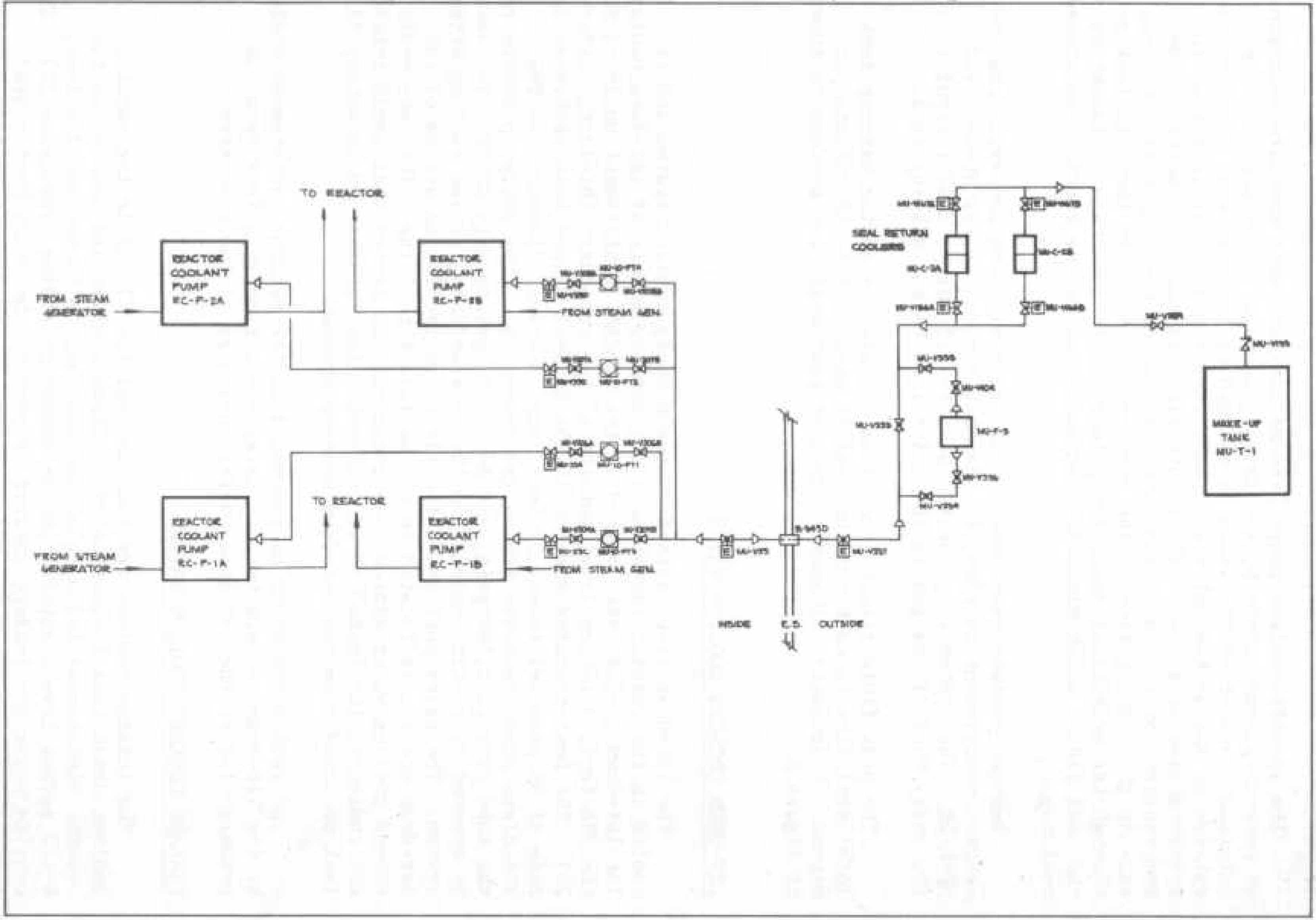
The let-down flow exits from the primary coolant system and is cooled in the reactor containment building by a pair of let-down coolers. The let-down coolers are located in the reactor containment building at the 286 feet, 3 inches level outside of the secondary shielding (reference 26). The heat-transfer surface in the helical-shaped heat exchanger is made of 30 parallel tubes, 57 feet long, 3/4-inch diameter, 16 BWG stainless steel seamless tubing (reference 19). The design pressure for the tube side is 2,500 psig, and design temperature is 600°F. The heat is removed by cooling water from the intermediate, closed cooling water system. The major portion of the closed recirculation system of the let-down coolers is located in the auxiliary building. The intermediate closed cooling water system is a recirculating system that would retain any radioactivity leaked to it and would also increase in inventory if a leak occurred from the let-down/make-up system.

For radioactivity to be removed from the reactor containment building by the let-down coolers cooling water would require a leak from the primary side of the let-down cooler tubing or of the tube sheet.

LEAKAGE COOLERS COOLING WATER

The leakage coolers are used to cool the fluids in the reactor coolant drain tank (reference 29). These coolers are located in the reactor containment building at the 285 feet, 6 inches and 290 feet, 8-1/2 inches levels outside the secondary shielding (reference 20). The cooling water for leakage coolers is part of the decay heat closed cooling water system. The major portion of the decay heat closed cooling

FIGURE 6: Reactor Coolant Pump Seals to Seal Coolers in Auxiliary Building



water is in the auxiliary building. Both the tube sides and the shell sides of the leakage coolers operate at approximately 5 to 10 psig. Therefore, there is no strong driving force to transmit radioactivity to the auxiliary building. via the leakage coolers cooling water.

VI. INTERCONNECTION OF LIQUID AND GAS SYSTEM

The liquid and gas systems are intercoupled, and communications can be established from one tank to another tank through an intermediate tank. Further, it is conceivable that the relief of pressure in one tank or system may, in turn, pressurize the recipient tank so that its relief valve setting may be exceeded.

In most systems there are check valves that function to allow flow in one direction only (reference 24). However, most check valves are normally not sufficiently reliable to prevent some backflow, particularly if the seats are metal. Relief valve vent systems do not normally have check valves since these systems are designed to be ultimate free paths of pressure relief.

During the accident, the major releases were gaseous. To assist in understanding the interconnections of the tanks, compressors, and other components, a summary drawing of the vent relief/waste gas ties is shown in Figure 7. Also, the overall connections of the most important systems considered in this report are shown in Figure 8.

In evaluating the systems at TMI-2, the reactor coolant bleed holdup tanks appear to be the center of the potential release pathways.

The reactor coolant drain tanks have liquid connections to the reactor coolant bleed holdup tanks as well as gas venting connections (reference 27). The let-down/make-up system is extensively tied into reactor coolant bleed holdup tanks (reference 6). The pressure relief valve MU-R3 just downstream from the block orifice valve discharges to the reactor coolant bleed holdup tanks. The pressure relief valve, MU-R1, on the outlet stream of the make-up tank discharges to the reactor coolant bleed holdup tanks. The flow from the let-down/make-up system can be diverted via a three-way valve, MU-U8, to the bleed holdup tanks. The waste transfer pumps WDL-P-5B and WDL-P-5A draw from the bleed holdup tanks and discharge into the line feeding into the let-down/make-up system just upstream of the make-up filters MU-F-2A and MU-F-2B.

The reactor coolant bleed holdup tanks are piped to the vent header in the auxiliary building, permitting ready venting of fluid from the tanks (reference 27). A line connects the auxiliary building sump tank to the vent gas header.

Another extensive set of interconnections is the relief valve vent system (reference 24). The relief valve vent system discharges directly to TMI-2 station vent. This is important because if a relief valve has lifted, it may not always reseat to be leak-free. This could very well have happened during or, in the case of the auxiliary building sump tank, before the accident. It is of interest to note that the relief valves from the waste gas compressors are piped to the auxiliary building sump tank that had a blown rupture disc at the time of the accident (reference 24).

The vent system is equipped with traps at the low points to drain water that might get into the vent system from lifting of pressure

FIGURE 7: Vent Gas/Relief Valve Header System

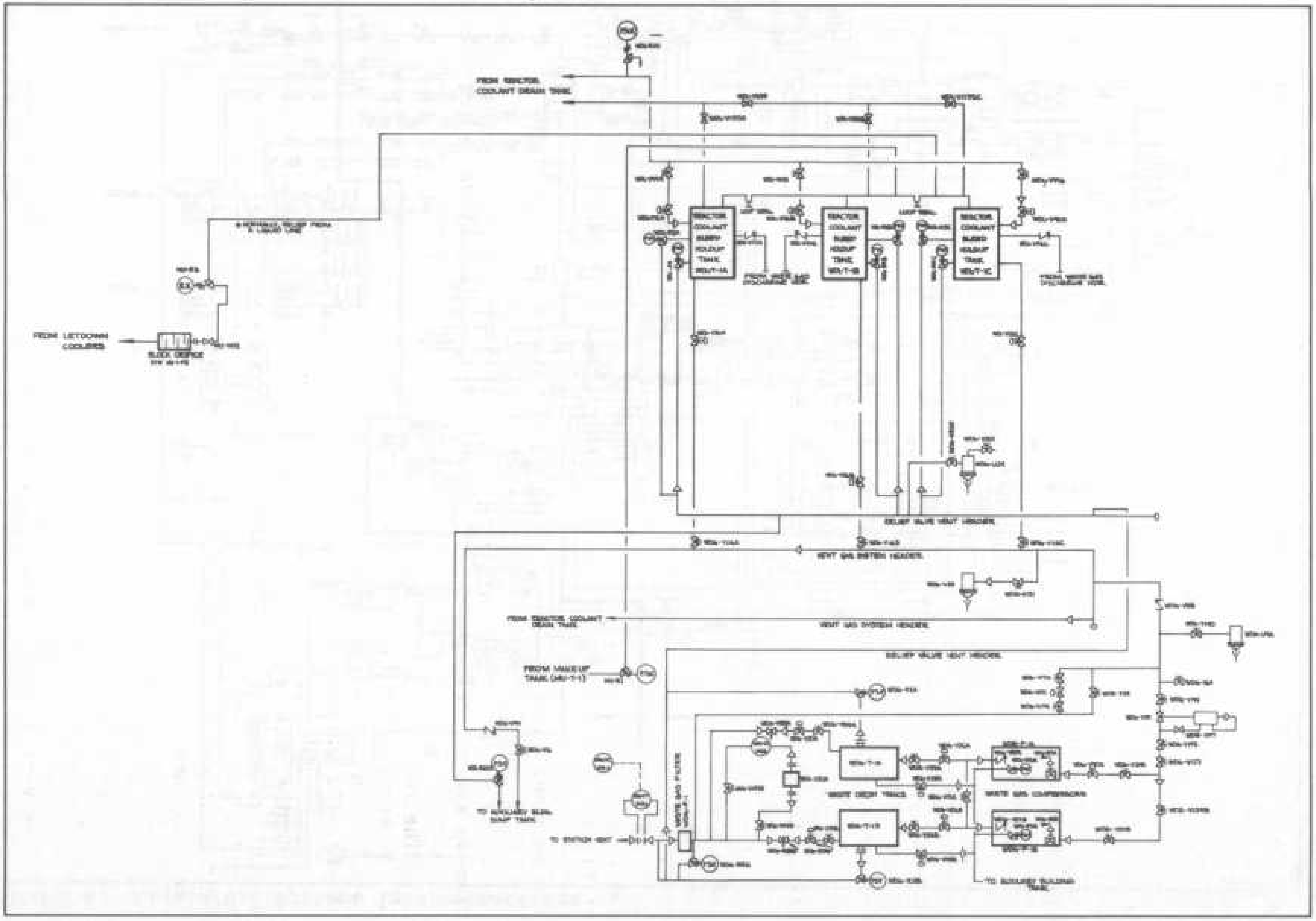
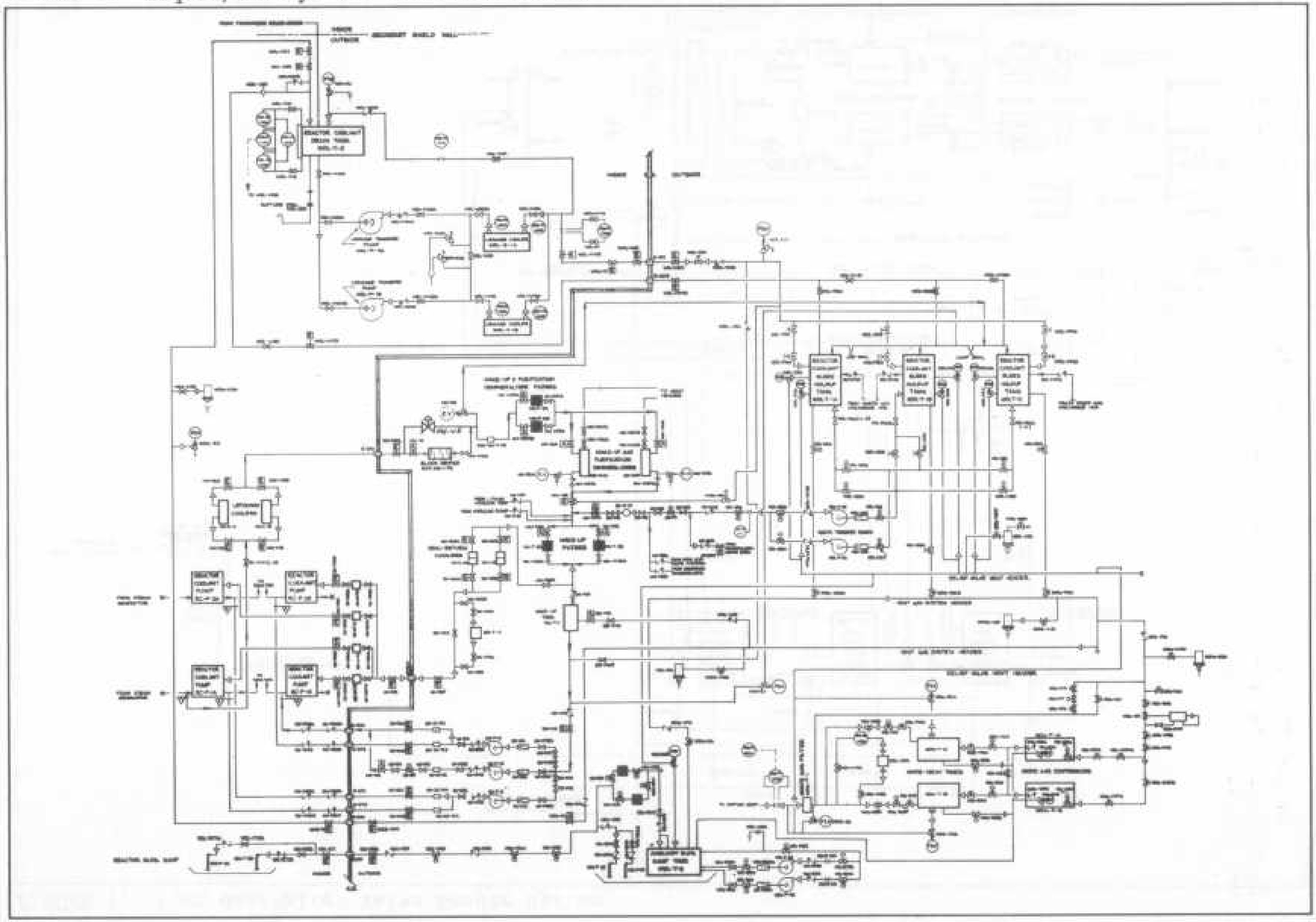


FIGURE 8: Liquid/Gas Systems Interconnections



relief valves. There are at least two of these traps, WDG-U8A and WDG-U9A, in the auxiliary buildings.

The interconnections of the liquid system with the gas systems make it difficult to diagnose the release points of the radioactive gas from the auxiliary building.

VII. RADIATION MONITORING IN THE PLANT

There are 49 radiation monitors in the TMI-2 plant (reference 5). Of these 21 are area monitors utilizing GM tubes for gamma level detection; 12 are airborne radioactivity monitors that measure particulate, iodine, and gaseous content; four are beta scintillation monitors for gaseous radioactivity; and 12 are gamma scintillation counters for detecting suspended or dissolved radioactive isotopes in liquid streams.

For purposes of this report, the only monitors that are useful are those for which data were retained on strip chart recorders. Listing of these recorded monitors are shown in Tables 1, 2, and 3. The response of each monitor is recorded by a printed number stamped on the recorder chart. Also included in Tables 1, 2, and 3 are the monitor locations as well as the drawing from which the locations were determined.

For more details, refer to the Met Ed/GPU "Preliminary Report on Sources and Pathways of TMI-2 Releases of Radioactive Material," Appendix G.

TABLE 1: List of Radiation Monitors and Locations Process Monitors

Monitoring Point	Recorder Stripchart	Recorder Number	Channel Number	Recorder Designator	Burns and Roe Drawing Number	Location on Drawing	Plant Elevation
Primary Coolant Let-down HL	HP-UR-3264	7	1	MU-R-720	2066	AE/A63.5	305'
Primary Coolant Let-down LO	HP-UR-3264	7	2	MU-R-720	2066	AE/A63.5	305'
Intermediate Coolant Let-down Cooler B	HP-UR-3264	7	3	IC-R-1091	2060	R11	282'6"
Intermediate Coolant Let-down Cooler A	HP-UR-3264	7	4	IC-R-1092	2060	R11	282'6"
Intermediate Coolant Let-down Cooler Outlet	HP-UR-3264	7	5	IC-R-1093	2066	AB/A62.8	305'
Plant Effluent Unit II	HP-UR-3264	7	6	WDL-R-1311			
Decay Heat Closed A Loop	HP-UR-3264	7	7	DC-R-3399	2065	AK/A67	280'6"
Decay Heat Closed B Loop	HP-UR-3264	7	8	DC-R-3400	2065	AK/A67	280'6"
Nucl. Serv. Closed Cooling	HP-UR-3264	7	9	DC-R-3401	2066	AE/A61	305'
Spent Fuel Cooling	HP-UR-3264	7	10	SF-R-3402	2066	AN/A64.7	305'

TABLE 2: List of Radiation Monitors and Locations Area Monitors

Area Monitored	Recorder Stripchart	Recorder Number	Channel Number	Recorder Designator	Burns and Roe Drawing Number	Location on Drawing	Plant Elevation
Control Room	HP-UR-1901	1	1	HP-R-201	2381	CB/C48	341'
Cable Room	RP-UR-1901	1	2	HP-R-202	2380	CC/C48	305'
Emerg. Cooling Booster Pump	HP-UR-1901	1	3	HP-R-204	2065	AB/A61	280'
R.C. Evap Control Panel Area	HP-UR-1901	1	4	HP-R-205	2065	AG/A63	280'
Make-up Tank Area	HP-UR-1901	1	5	HP-R-206	2066	AG/A64	305'
Intermediate Cooling Pump Area	HP-UR-1901	1	6	HP-R-207	2066	AA/A63	305'
Fuel Handling Bridge North	HP-UR-1901	1	7	HP-R-208	2064		355'
Fuel Handling Bridge South	HP-UR-1901	1	8	HP-R-209	2064		355'
R. B. Personnel Access Hatch	HP-UR-1901	1	9	HP-R-210	2064		310'
R. B. Equipment Hatch	HP-UR-1901	1	10	HP-R-211	2064		314'
Incore Instrument Panel Area	HP-UR-1901	1	11	HP-R-212	2062		371'
Reactor Building Dome	HP-UR-1901	1	12	HP-R-213	2064		372'
Fuel Handling Bridge	HP-UR-1902	2	1	HP-R-215	2068	AM/A66.5	347'
Waste Disposal Storage Area	HP-UR-1902	2	2	HP-R-218	2066	AR/A67	305'
Aux Bldg Sump Tank Filter Room	HP-UR-1902	2	3	HP-R-231	2065	AQ/A62	280'
Aux Bldg Access Corridor	HP-UR-1902	2	4	HP-R-232	2066	AR/A61	305'
Aux Bldg Access Corridor	HP-UR-1902	2	5	HP-R-233	2066	AN/A63	305'
Control & Service Bldg	HP-UR-1902	2	6	HP-R-234	2380	CE/C50A	280'
RB Purge Unit Area	HP-UR-1902	2	7	HP-R-3236	2067	AE/A63	328'
Aux Bldg Exh. Unit Area	HP-UR-1902	2	8	HP-R-3238	2067	AJ/A63	328'
Fuel Handling Exh. Unit Area	HP-UR-1902	2	9	HP-R-3240	2067	AL/A63	328'

TABLE 3: List of Radiation Monitors and Locations Airborne Monitors

Monitoring Point	Recorder Stripchart	Recorder Number	Channel Number	Recorder Designator	Burns and Roe Drawing Number	Location on Drawing	Plant Elevation	Monitored Variable
Station Vent	HP-UR-1907	4	1	HP-R-219	2067	AF/A65	328'	Particulate
Station Vent	HP-UR-1907	4	2	HP-R-219	2067	AF/A65	328'	Iodine
Station Vent	HP-UR-1907	4	3	HP-R-219	2067	AF/A65	328	Gas
Control Room Intake	HP-UR-1907	4	4	HP-R-220	2381	CD/C48	351'6"	Particulate
Control Room Intake	HP-UR-1907	4	5	HP-R-220	2381	CD/C48	351'6"	Iodine
Control Room Intake	HP-UR-1907	4	6	HP-R-220	2381	CD/C48	351'6"	Gas
Fuel Handling Bldg Exh. Upstream of Filter	HP-UR-1907	4	7	HP-R-221A	2067	AT/A63	328'	Particulate
Fuel Handling Bldg Exh. Upstream of Filter	HP-UR-1907	4	8	HP-R-221A	2067	AT/A63	328'	Iodine
Fuel Handling Bldg Exh. Upstream of Filter	HP-UR-1907	4	9	HP-R-221A	2067	AT/A63	328'	Gas
Fuel Handling Bldg Exh. Downstream of Filter	HP-UR-1907	4	10	HP-R-221B	2067	AT/A63	328'	Particulate
Fuel Handling Bldg Exh. Downstream of Filter	HP-UR-1907	4	11	HP-R-221B	2067	AT/A63	328'	Iodine
Fuel Handling Bldg Exh. Downstream of Filter	HP-UR-1907	4	12	HP-R-221B	2067	AT/A63	328'	Gas
Hydrogen Purge	HP-UR-1907	4	13	HP-R-229	2067	AF/A65	328'	Particulate
Hydrogen Purge	HP-UR-1907	4	14	HP-R-229	2067	AF/A65	328'	Iodine
Hydrogen Purge	HP-UR-1907	4	15	HP-R-229	2067	AF/A65	328'	Gas

TABLE 3: List of Radiation Monitors and Locations Airborne Monitors (continued)

Monitoring Point	Recorder Stripchart	Recorder Number	Channel Number	Recorder Designator	Burns and Roe Drawing Number	Location on Drawing	Plant Elevation	Monitored Variable
RB Purge Air Exh. Duct A	HP-UR-2900	5	1	HP-R-225	2067	AB/A64	328'	Particulate
RB Purge Air Exh. Duct A	HP-UR-2900	5	2	HP-R-225	2067	AB/A64	328'	Iodine
RB Purge Air Exh. Duct A	HP-UR-2900	5	3	HP-R-225	2067	AB/A64	328'	Gas
RB Purge Air Exh. Duct B	HP-UR-2900	5	4	HP-R-226	2067	AB/A64.5	328'	Particulate
RB Purge Air Exh. Duct B	HP-UR-2900	5	5	HP-R-226	2067	AB/A64.5	328'	Iodine
RB Purge Air Exh. Duct B	HP-UR-2900	5	6	HP-R-226	2067	AB/A64.5	328'	Gas
Aux Bldg Purge Air Exh. Upstream of Filter	HP-UR-2900	5	7	HP-R-222	2067	AT/A63	328'	Particulate
Aux Bldg Purge Air Exh. Upstream of Filter	HP-UR-2900	5	8	HP-R-222	2067	AT/A63	328'	Iodine
Aux Bldg Purge Air Exh. Upstream of Filter	HP-UR-2900	5	9	HP-R-222	2067	AT/A63	328'	Gas
Aux Bldg Purge Air Exh. Downstream of Filter	HP-UR-2900	5	10	HP-R-228	2067	AT/A63	328'	Particualte
Aux Bldg Purge Air Exh. Downstream of Filter	HP-UR-2900	5	11	HP-R-228	2067	AT/A63	328'	Iodine
Aux Bldg Purge Air Exh. Downstream of Filter	HP-UR-2900	5	12	HP-R-228	2067	AT/A63	328'	Gas
Reactor Bldg Air Sample	HP-UR-3236	6	1	HP-R-227	2066	AB/A63	328'	Particulate
Reactor Bldg Air Sample	HP-UR-3236	6	2	HP-R-227	2066	AB/A63	328'	Iodine
Reactor Bldg Air Sample	HP-UR-3236	6	3	HP-R-227	**2066**	AB/A63	328'	Gas
Waste Gas Discharge Duct	HP-UR-3236	6	4	WGD-R-1480	2067	AB/A62.5	328'	Gas
WDG-T-1A Waste Gas Decay Tank Discharge	HP-UR-3236	6	5	WGD-R-1485	2066	AG/A62.5	305'	Gas
WDG-T-1B Waste Gas Decay Tank Discharge	HP-UR-3236	6	6	WGD-R-1486	2066	AD/A62.5	305'	Gas
Condenser Vacuum Pump Discharge	HP-UR-3236	6	7	VA-R-748	2052	TG/T42	281'6"	Gas

VIII. POTENTIAL PRESSURE TRANSIENTS IN EACH SYSTEM

LET-DOWN/MAKE-UP SYSTEM

The let-down/make-up system is explained in Section V. This system normally is the main flow stream of reactor primary coolant from the reactor containment building.

Pressure in the reactor coolant system is normally imposed and sustained by the pressurizer through the use of heaters (reference 17). There is a slight partial pressure of hydrogen maintained to assure recombination of hydrogen and oxygen produced by radiolysis.

Since there is a minimum of dissolved gases in the primary coolant, the let-down/make-up system is designed to handle a modest amount of gases by venting into the waste gas vent header from the make-up tank.

The block orifice valve MU-1-FE in the let-down flow line is located in the auxiliary building (reference 6). The valve is designed to reduce the pressure in the system from a system pressure of 2,135 psig at the outlet of the let-down cooler to about 20 psig (reference 29). The relief valve MU-R-3 protects the downstream components from over-pressure.

During the accident, three circumstances could, and probably did, cause the relief valve MU-R-3 to lift: (1) When the bypass valve MU-V5 around the block orifice was opened to increase let-down flow, there is a potential of a mismatch in flow-related pressure drop through the make-up and purification demineralizer filters, the make-up and purification demineralizers and the associated piping before the make-up tank (references 30, 31, 32); (2) If there is a flow blockage downstream of the block orifice valve when the let-down system is functioning, the pressure up to the point of blockage would rise until the relief valve lifted; and (3) When there are sufficient dissolved gases in the primary system, the reduction of pressure through the block orifice valve can cause substantial degassing with subsequent two-phase flow that changes the pressure drop through the filters and demineralizers. There is a pressure indicator, MU-Pl-1579, just downstream of the block orifice valve. However, since there is no recorder, no trace is available for analysis.

A review of the operations during the accident indicates that all three circumstances were probably experienced (references 30, 31, 32). The relief valve is set at 130 psig and discharges into the reactor coolant bleed holdup tanks (references 33, 34, 35). A mixture of liquid and gases would be the discharge phases from the relief valve after 6:30 a.m., on March 28 (references 36, 37, 38). This is the approximate time when substantial amounts of hydrogen were generated in the primary system by the zircaloy-water reaction.

REACTOR BUILDING SUMP TO THE AUXILIARY BUILDING SUMP

Flow of primary water from the reactor building sump to the auxiliary building sump occurred for approximately 30 minutes during the initial 38 minutes of the accident when the reactor building sump pumps were operating (references 1, 2, 5). The Met Ed/GPU analysis concluded that siphon flow probably did not occur with the pumps shut down. This writer reviewed the plant drawings and agrees with the analysis conclusion.

Calculation by Met Ed/GPU indicates a significant amount of water was pumped by the reactor building sump pump (8,400 gallons) (reference 5). An assumption was made that the water flowed into the auxiliary building sump tank and then into the sump via the burst rupture disc WDL-U224. In examining the overall flow system, some of the water could very well have gone up the vent line into the vent header, depending on the res pective flow resistance and the vent header pressure during the pumping period. The injection of water into the vent gas header would have deleterious effects on the operation of the vent gas header system.

There is some confusion as to the status of the controls for the reactor building sump pumps. They were turned off at about 6:38 a.m., and an auxiliary building operator recalled they were not operating at about 7:00 a.m.; however, a shift supervisor reported he found the reactor building sump pump controls in the automatic position at about 1,200-1,300 and turned them off (reference 39).

If the pumps did run for the period from 7:00 a.m., until 7:56 a.m., when the reactor containment building was isolated by a pressure signal, another 7,800 gallons may have been pumped into the auxiliary building sump. The inlet of the reactor building sump pumps is at the bottom of the sump and would probably have continued to pump "first out" primary water that was low in radioactivity. Whatever water was pumped out by the reactor building sump pumps would have been degassed to a large extent by steam stripping and depressurization as the water exiting from the electromatic relief valve through the reactor collant drain tank.

In reviewing the overall behavior of circumstances of pumping water from the reactor building sump to the auxiliary building sump via the auxiliary building sump tank, it is not obvious that this caused any pressure transients, but it did contribute to the inventory of initial primary system water in the auxiliary building.

REACTOR COOLANT DRAIN TANK TO THE REACTOR COOLANT BLEED HOLDUP TANKS

A review of operations before the accident indicated that the reactor coolant drain tank received leakage from the pressurizer relief valves (probably the safety relief valves based on temperatures) (references 28, 4). The chart from flow recorder FR 7100 indicated that periodically leakage water from the reactor coolant drain tank was being transferred to the reactor coolant bleed holdup tank through the reactor coolant drain header. To perform this transfer, one of the leakage transfer pumps, WDL-P-9A or WDL-P-9B, must be operating. Since there were periodic transfers prior to the accident, it is believed that the

The transfer of liquid from the reactor coolant drain tank to the reactor coolant bleed holdup tanks requires the opening of valve WDL-V1118 by use of remote controls (reference 28). Level indicator and controller WDL-1206 will automatically close WDL-V1118, if the level in the tank goes below 72 inches.

Probably the best indication of flow to the reactor coolant bleed holdup tanks is the flow recorder FR 7100. The flow range on the chart is 150 gallons per minute maximum.

There should have been no flow after the reactor containment building was isolated at 7:56 a.m. This statement is inconsistent with the indi cation on the chart from the recorder (reference 4). The anomalies are that there were extensive flow indications from about 9:08 a.m. until 2:30 p.m., i.e., the start of what one might expect from the output of this record was 9:08 a.m. instead of 4:00 a.m., and the duration was 5 hours, 22 minutes instead of 3 hours, 56 minutes.

If one assumes the chart speed is not correct and the stamped date is in error by approximately 5 hours, the traces on the chart are quite plausible. There were four periods of flow for about 50 percent of the lapse time which corresponds quite well with the open electromatic relief block valve plus a period with the block valve being closed and then reopened. The chart indicated intermittent flows greater than 150 gallons per minute. By normalizing the flow periods to the total lapse time of 3 hours, 56 minutes, a total transfer of 9,450 gallons could have occurred. This would not have exceeded the capacity of the reactor coolant bleed holdup tanks. The liquids also would have been essentially degassed during the flow from the pressurizer to the reactor coolant drain tank.

REACTOR COOLANT DRAIN TANK VENT TO VENT GAS HEADER IN THE AUXILIARY BUILDING

The reactor coolant drain tank vent is connected to the reactor building vent header by approximately 70 feet of one-inch pipe (references 40, 41). The 4-inch reactor building vent header is coupled to the auxiliary building vent gas system. During normal operation, venting would be modest due to relatively slow changing levels in the reactor coolant drain tank.

When the electromatic relief valve opened at the time of the accident, the reactor coolant drain tank was pressurized first up to 120 psig when the pressure relief valve opened, and then on up to 190 psig when the rupture diaphragm burst at about 15 minutes into the accident (reference 42). The tank was subjected to pressures between 120 psig and 190 psig for some 12 minutes.

During this 12-minute period, a mixture of mostly steam and water was forced into the reactor vent header and then into the auxiliary building vent system. Some of the water was removed by the trap WDG-U10A in the reactor containment building (reference 24). The steam was condensed in the piping. A reasonable assumption would be that most of the fluid that went out the vent was water. If one assumed a pressure differential of 100 psi across the one-inch pipe in 12 minutes, over 1,000 gallons of water may have been ejected from the reactor coolant drain tank into the auxiliary building vent header. This appears to be an excessive amount of water. It also could be assumed that the release fluids through the vent line were a mixture of steam and water that would reduce the condensed volume of transported liquids (reference 43). To give a perspective, 1,000 feet of 4-inch diameter schedule 40 pipe has an internal volume of 660 gallons.

From the above, it can be seen that the vent headers in both the reactor building and the auxiliary could have received a substantial amount of water if the valves in the vent system from the reactor coolant drain tank were lined up normally.

The water would deteriorate the capabilities of the auxiliary building vent header (reference 44). However, none of this water would have had extensive radioactivity since this event occurred prior to the uncovering of the core.

If the auxiliary building vent header had any leaks at the inception of the accident, this pathway would take even minor amounts of radioactive gas and liquids directly over to the auxiliary building (reference 7). This might explain some of the early air-borne radiation monitor responses (see Section VII, Table 2).

The outlet of the 18-inch diameter vent pipe housing the rupture disc is 7 feet above the reactor coolant drain tank. If one assumes a liquid head of water equivalent to about 3 psi, the internal pressure for liquid or gases after the failure of the rupture diaphragm would be 4.5 to 5 psig. Depending on the pressure in the auxiliary building vent header, flow may or may not have occurred after the rupture diaphragm failure.

It was reported that the vent valves WDL-U126 and WDL-U127 were found open after the accident and subsequently closed on June 5, 1979, (reference 5).

REACTOR COOLANT DRAIN TANK VENT TO REACTOR COOLANT BLEED HOLDUP TANKS

The reactor coolant drain tank communicates with the reactor coolant bleed holdup tanks through a vent line also. This vent line is automatically closed by valve WDL-1095 when the pressure sensor WDL-1203 signal exceeds 10 psig (reference 46). During the accident, the valve should have closed within 2 minutes and remained closed until the rupture diaphragm burst at 15 minutes.

After the rupture diaphragm burst, the driving pressure would have been about 4.5 to 5 psi until isolation occurred at 7:56 a.m. After isolation was effected, the driving pressure was purely academic since no flow could result.

It is questionable that during the initial part of the accident -- the first 2 minutes -- that the vent line could have been filled with liquid to set up a potential siphon. A siphon is also unlikely since the reactor coolant bleed holdup tanks are connected to the auxiliary building vent header and therefore do not have free liquid surfaces exposed to the atmosphere. Further, it is likely that pressure built up as liquid was routed to the reactor coolant bleed holdup during the course of the accident.

REACTOR COOLANT PUMP SEALS TO THE SEAL RETURN COOLERS IN THE AUXILIARY BUILDING

The reactor coolant pump seal water to the seal return coolers in the auxiliary building are fundamentally part of the reactor coolant let-down/make-up system. The pump seal water is furnished by the output of make-up pumps, normally pump MU-P-1B. The seal water may become contaminated by the primary coolant if the primary coolant pump mechanical seals are leaking.

The effluent pump seal water is cooled and returned to the reactor coolant let-down/make-up system just upstream of the make-up tank.

The reactor primary coolant pumps were subjected to rather severe conditions during the accident (reference 3). These severe conditions which included vibration and pumping of two-phase fluids could have damaged the mechanical seals and allowed primary coolant to mix with the seal fluid. The seal water return could therefore have transmitted radioactivity to the make-up tank.

The behavior of the reactor coolant pump seal water being returned to the let-down/make-up system appeared normal, without incident, throughout the accident.

LET-DOWN COOLERS COOLING WATER

A review of the records of the intermediate cooling water system did not reveal any abnormalities in the operation utilizing the let-down coolers cooling water (reference 3).

LEAKAGE COOLERS COOLING WATER

A review of the records of the operation of the leakage coolers did not reveal any abnormalities in the function of the cooling water (refer ence 3).

INTERCONNECTION PRESSURE TRANSIENTS

The relief valve MU-R-3 from the let-down/make-up system downstream of the block orifice valve MU-1-FE discharges into the reactor coolant bleed holdup tanks. It is believed that the discharges from the relief valve contained gas and liquid and subsequently increased the pressure in the bleed holdup tanks.

The release of dissolved gases can sometimes take appreciable time. For this reason, the gas phase and the liquid phase discharging from the block orifice valve may not be in equilibrium. The equilibrium may not be obtained until the phases reach a tank where there is sufficient residence time. The make-up tank serves this purpose. A review of the records shows that the make-up tank became pressurized and lifted the relief valve MU-RI downstream of the tank (reference 35). This relief valve discharges into the reactor coolant bleed holdup tanks, also increasing the pressure in these tanks.

The combined pressure increases could and did lift the relief valves of the reactor coolant bleed holdup tanks (references 35, 36, 37). The relief valves from the reactor coolant bleed holdup tanks discharge to the relief valve vent header that in turn discharges to the station vent (reference 24).

IX. RESPONSE OF RADIATION MONITORS DURING THE ACCIDENT

The radiation monitoring system at the TMI-2 plant is designed primarily for operating conditions with most setpoints in the low range, 20 millirems per hour or lower, (reference 5). Areas where it is reasonably certain that high radiation levels may be encountered during fuel handling operations, the range of the equipment is higher. The locations of the monitors with recorders are presented in Section VII.

The strip charts from the radiation monitor recorders give the most significant information with respect to where and when radioactivity was detected in the plant. There are, however, certain limitations. Each monitor response is identified by a number stamped on the recorder chart. In many cases, the stamped numbers were not inking properly at the time of the accident. This makes precise reading of the monitoring data very difficult, and, in some instances, it is not possible to differentiate between different data points. The diagnostic readings are further complicated by imprecise timing marks on the strip charts. However, within a strip chart, relative times are quite identifiable. Comparison between charts, inserts, probable errors, and care must be exercised in interpretation. In summary, it is the best time information available and should be used fully realizing the limitation.

A compilation of the early radiation monitoring responses is shown in Table 4 (references 7, 9, 10, 11). As in past experience in nearly all reactors, when a scram occurs there is a slight increase in radioactivity distributed within the plant primary system, due to two reasons: (1) There is normally a mismatch between heat production in the core and coolant flow, and (2) There is a rapid cooling of the fuel pins that creates some stress in the cladding.

The mismatch in overheat production and coolant flow can dislodge small radioactive particles that have accumulated, thereby creating a "crud" shower. The crud shower is normally confined within the reactor coolant system so that no gaseous evolution occurs.

The rapid cooling of the fuel pins can create stress, and if there is marginal cladding in any of the 39,000 pins in the reactor core, a defect could occur, and radioactivity might be released to the coolant. These fission products would be noble gases and volatile fission products.

It is believed that the initial responses to the radiation monitors reflect the rupturing of some fuel pins in the reactor core at or shortly after reactor power was shut down. The fact that monitors in the auxiliary building indicated radioactivity shows that the gas systems were not leak-tight at the inception of the accident. It should be realized that it is almost a Herculean task to ever get a reactor plant with the numerous pumps and valves to be a completely leak tight system.

In the case of TMI-2, the movement of primary water from the reactor containment building sump to the auxiliary building sump via the failed rupture disc U-224 of the auxiliary sump tank during the initial 38 minutes probably prolonged the initial minor release. This probably

TABLE 4: Early Radiation Monitoring Responses, March 28, 1979

Recorder	Instrument Number	Instrument Description	Starting Time of Increase			
2900 (6)*	HP-R-226-G	R.B. Purge Air Exhaust, Duct A	4:07	4:22	6:43	
2900 (3)	HP-R-225-G	R.B. Purge Air Exhaust, Duct B	4:07	4:22		
1907 (14)	HP-R-229-I	Hydrogen Purge	4:15* See note below			
3236 (3)	HP-R-227-G	R.B. Air Sample	4:16	6:22		
3236 (1)	HP-R-227-P	R.B. Air Sample	4:16	6:25		
3236 (2)	HP-R-227-I	R.B. Air Sample	4:16	6:31		
1907 (3)	HP-R-219-G	Station Vent	4:19**			
1907 (7)	HP-R-221A-P	F.H.B. Exhaust Upstream of Filters	4:22**			
1907 (10)	HP-R-221E-P	F.H.B. Exhaust Downstream of Filters	4:22**			
3264 (4)	IC-R-1092	Intermediate Cooling of Let-down Cooler A	4:22	5:00	5:23	6:37
2900 (9)	HP-R-222-G	A.B. Purge Air Exhaust Upstream of Filters	4:28			

*Instrument code number.

**These recorders' data are difficult to separate due to illegible traces. They appear to essentially respond together and rose slightly at the 4:15 to 4:20 time period and then rose again at 6:40 to 6:50.

was the source of the first response of the monitors in the auxiliary building. Obviously the responses in the reactor containment building were from venting of the reactor coolant drain tank by way of the failed rupture disc.

The responses of IC-R-1092 in which there were increased boils at 5:00 a.m. and 5:23 a.m. were probably caused by more direct exposure to the liquid in the reactor building sump which is just below the location of this monitor.

The major releases of radioactivity started just 2 hours into the accident when the core became uncovered. It appeared that the first instrument to indicate additional and increasing radioactivity was the reactor containment building air sampler HP-R227 at approximately 6:22 a.m. This would indicate there was a delay of 10 to 20 minutes from the time that it was thought the first uncovering of the core until fission gases got outside the reactor primary system.

Of the area monitors, the reactor containment building monitors, HP-R-213, -214, -209, and -210 first gave increased responses starting at about 6:31 a.m. (reference 12). In the auxiliary building, HP-R-207 showed a modest increase at about 6:41 a.m. and a sharp increase at 7:19 a.m. This first response of HP-R-207 at 6:41 a.m. may have been from the reactor containment building since this monitor is right against the containment building. These and other responses are shown in Table 5.

Coupling the response of HPR207 at 6:41 a.m. with the make-up tank area monitor HPR206 at 6:43 a.m. and intermediate coolant let-down cooler outlet monitor ICR1093 at 6:43 a.m., it appears that the initial radioactivity reached the auxiliary building about 10 minutes after at least some dispersion in the reactor containment building (reference 12). This time delay is quite logical, considering that in the case of let-down flow, the radioactivity had to get from the core over to the let-down line outlet in the A loop of the reactor coolant system.

Examination of the strip chart from HP-UR-1902 during the period early March 30 at 1:35 a.m. and 3:33 a.m. showed a difference in the response of radiation monitors HP-R-208, -232, -3236, and -3240 (reference 13). When the make-up tank was vented to the vent header, there appeared to be a delay of 4 to 10 minutes before the monitors showed an increase in radioactivity. Monitors HP-R-3240 and -3236 responded concurrently but to a different extent. Both of these monitors are on the 328-foot level of the auxiliary building. Monitor HP-R-232 in the access corridor near the radwaste panel area on the 305-foot level responded about 2 to 5 minutes after the above monitors indicated increased radiation. HP-R-232 response usually was greater than HP-P-3236 but less than HP-R-3240. HP-R-218 was at the 305-foot level in the fuel handling building waste disposal area and showed increased radiation about 7 to 15 minutes after HP-R-232. The HP-R-218 response did not peak as much as the others and probably showed just the overall fuel building/auxiliary building background change. The write-up by Met Ed/GPU indicates that HP-R-207 (305 foot level) and HP-R-204 (208 foot

TABLE 5: Other Radiation Monitoring Responses, March 28, 1979

Recorder		Instrument Number	Instrument Description	Starting Time of Increase		
1901	(11)	HP-R-213	Incore Inst Panel Area	6:31	a.m.	
1901	(12)	HP-R-214	R.B. Dome	6:32		
1901	(7)	HP-R-209	F.H. Bridge, N	6:33		
1901	(8)	HP-R-210	F.H. Bridge, S	6:34		
2900	(10)	HP-R-228-P	A.B. Purge Air Exh Downstream of Filters	6:40		
1901	(6)	HP-R-207	Intermediate Cooling Pump Area	6:41	7:19	
2900	(4)	HP-R-226-P	R.B. Purge Air Exh Duct B	6:42		
2900	(1)	HP-R-225-P	R.B. Purge Air Exh Duct A	6:42		
3236	(7)	VA-R-748	Condenser Vacuum Pump Discharge	6:42		
1901	(5)	HP-R-206	Make-up Tank Area	6:43		
3264	(5)	IC-R-1093	Intermed Coolant Let-down Cooler Outlet	6:43		
2900	(12)	HP-R-228-G	A.B. Purge Air Exh Downstream of Filters	6:45		
1902	(2)	HP-R-218	Waste Disposal Storage Area	6:46		
1902	(7)	HP-R-3236	R.B. Purge Unit Area	6:48		
3264	(9)	NS-R-3901	Nucl Service Closed Cooling	6:53		
3236	(4)	WDG-R-1480	Waste Gas Discharge Duct	6:54		
1902	(1)	HP-R-215	F. H. Bridge	6:55	7:45	9:37
1902	(9)	HP-R-3250	F. H. Exh Unit Area	6:58	7:22	
3264	(1)	MU-R-720	Hi Primary Coolant Let-down	7:04		
3264	(2)	MU-R-720	Lo Primary Coolant Let-down	7:04		
3236	(6)	WDG-R-1486	Waste Gas Decay Tank B Discharge	7:15		
1902	(4)	HP-R-232	A.B. Access Corridor	7:21		
3264	(6)	WDL-R-1311	Plant Effluent TMI-2	7:26		
3264	(10)	SF-R-3402	Spent Fuel Cooling	7:29		
3264	(8)	DC-R-3399	Decay Heat Closed A Loop	7:45		
3264	(9)	DC-R-3400	Decay Heat Closed B Loop	7:45		
3236	(5)	WDG-R-1485	Waste Gas Decay Tank A Discharge	7:52		
1902	(6)	HP-R-234	Control & Service Bldg Access Corridor	7:55	9:46	

level) both at the other end of the auxiliary building from HP-R-232 responded similarly to HP-R-218.

In diagnosing these responses, it appears that the major portion of the released gas went out of the vent header and into the ventilation system that was monitored by HP-R-3240 and HP-R-3236. However, there also appears to be a path opening somewhere near HP-R-232 that took longer to reach HP-R-232 than for the gas to get into the ventilation system and out the station vent. This would indicate quite strongly that there is more than one escape point of the gas from the vent header or associated systems.

On April 1 there was an effort to vent the waste gas decay tanks to the reactor containment building. The line was to be connected from radiation monitor WDG-R-1486 on the outlet of waste gas decay tank WDG-T-1B through a flame arrester and into the reactor containment building through penetration R-571C. After four attempts in which various leaks were found and repaired or isolated, a tight system was obtained. The rudiments of this effort are explained in Appendix C of the Met Ed/GPU "Preliminary Report on Sources and Pathways of TMI-2 Release of Radioactive Material and SOP Z2 (reference 46). Radiation monitor WDG-R-1486 is on level 305 feet of the auxiliary building.

During the initial attempts to utilize the venting system, leaks were found at WDG-R-1486 and WDG-R-1485. The gas to be vented was probably the most concentrated radioactive gas in the system exclusive of the make-up tank and the primary coolant system.

A review of strip chart of April 1 at 4:31 a.m. and 6:30 a.m., when waste gas valve WDG-V-30B was opened showed that HP-R-3240 and HP-R-3236 responded within several minutes (2-3). HPR232 responded only a minute or two after HP-R-3240 and HP-R-3236. HP-R-218 was responding at about another 5 minutes afterward.

These responses are just about what would be expected for an open release in the auxiliary building. The ventilation system immediately starts exhausting the radioactive gas and hence the relatively quick response of HP-R-3240 and HP-R-3236. HP-R-232 which is some 100 to 150 feet away would probably see the cloud due to diffusion through the auxiliary building (references 13, 47). HP-R-218 is even more isolated by being over against the opposite side of the fuel handling building and should probably see it well after HP-R-232.

Since these were known to be releases directly to the atmosphere in the auxiliary building, the response times with respect to the venting of the make-up tank to the radwaste vent header showed that the released radioactivity from the make-up tank had to follow some torturous path before exiting the system.

During early March 29, there were indications on the HP-UR-1907 strip chart that the ventilation was secured from about one hour 5 minutes (1:05 to 2:10 a.m.). The radiation monitors started rising immediately. The various monitor readings went up by the following factors:

HP-R-3236 - 2	HP-R-218 - 20
HP-R-3240 - Off-scale & then a factor of 5	HP-R-234 - 60
	HP-R-215 - 400
HP-R-232 - 8	

After the ventilation was restarted at 2:10 a.m., March 29, the monitoring readings started to decrease. This indicated that radiation levels within the auxiliary building were being kept lower by continued exhausting of auxiliary building air.

X. DISCUSSION AND RATING OF POTENTIAL PATHWAYS

The release of airborne radioactivity from the TMI-2 was not a single episode but a series of events that were unavoidable given the design of the plant. The analyses of the fission products that escaped from the reactor core indicated that they had been processed through water prior to being released. The major isotopes were xenons, krypton, and some iodine. The releases were somewhat complex in that several events occurred before the releases were made.

The releases, for the most part, were involuntary, but several were specifically made in attempts to gain better control of the plant.

The pathways primarily considered in this report are those that had high potential of significant leakage or radioactivity. The major characteristic of these potential pathways was the proximity to the reactor primary coolant which was under pressure. It should be understood that had the circumstances and results of the accident been different, a different evaluation would have been performed to address that situation.

It is believed that events during the first hour of the accident set up some of the conditions contributing to the uncontrolled release of radioactivity. The destruction of the core completed the setup.

The pressurization of the reactor coolant drain tank is believed to have caused water to be forced into the auxiliary building vent header, probably at some elevated pressure. This pressure is believed to have damaged the internals of the liquid traps of the vent header, thereby setting up one of the leakage pathways. The water could have damaged such other components as valves and the operation of the waste gas compressors. The likely release points of the radioactivity escapes, evolved with venting the make-up tank to the vent header, were the damaged traps or other components.

The relief valves of the waste gas compressor vent into the auxiliary building sump tank. If there had been major releases early in the accident (less than 5 hours) due to lifting of these relief valves, it would be expected that radiation monitor 232, which is the closest monitor, would have responded. This monitor did not exhibit the expected response which would have been expected had the release been from the auxiliary building sump tank.

The reactor coolant bleed holdup tanks are the recipients of the relief valves of the make-up tank and of downstream of the block orifice valve. The routing of let-down flow, via the three-way valve MU-V8, into the reactor coolant bleed holdup tanks on March 29, also increased the pressure in these tanks. The lifting of relief valves from the reactor coolant bleed holdup tanks discharges fluid into the relief valve vent header that is connected, unencumbered, to the station vent.

The attempts and final success of venting the waste decay tanks to the reactor containment building were traceable and present essentially no uncertainties as to pathways and time of release.

The analyses of the waste gas decay tanks indicate high hydrogen content, which substantiates the thesis that the radioactive gases were carried along by the degassing process.

There were some very low radioactivity-content water releases as explained by the NRC. Since the release was so small, little is said about liquid releases in this report.

A rating of pathways can be made by examining data from the operations and releases. This provides a reasonable ranking of pathways. In the order of importance, the following appears consistent with the data:

- Reactor coolant let-down/make-up system
- Reactor coolant drain tank vent to the vent header in the auxiliary building
- Reactor coolant drain tank to reactor coolant bleed holdup tanks
- Reactor coolant drain tank vent to the reactor coolant bleed holdup tanks
- Reactor building sump to auxiliary building sump

The other pathways appear to be negligible with respect to the above list.

XI. MOST PROBABLE PATHWAY

The most probable pathway is the let-down/make-up system. The venting of the reactor coolant bleed holdup tanks and make-up tank to the vent header and the pressure relief valve liftings of the reactor coolant bleed holdup tanks are considered to be the major pathways of uncontrolled radioactive releases. All other pathways basically terminate when isolation of the reactor containment building occurs.

During the first 3 hours, 56 minutes, some release may have been through the reactor coolant drain tank vent to the vent gas header in the auxiliary building. However, the transfer of water to the vent with the pressurizing effects on the vent header is believed to be the most important aspect of this pathway.

In summary, the fission products were released to the reactor primary coolant system and exited the primary system through the let-down/make-up system into the auxiliary building. The fission products were released through the damaged vent header to the auxiliary building and the fuel handling building and from the lifting of the pressure relief valves of the reactor coolant bleed holdup tanks that vent directly to the environs through the station vent. From the analyses of the charcoal from the auxiliary building exhaust and the fuel handling building exhaust, a major portion of the airborne activity release in these buildings was exhausted through the fuel handling building exhaust.

APPENDIX A

DISCUSSION OF METROPOLITAN EDISON/GENERAL PUBLIC UTILITIES CORPORATION'S PRELIMINARY REPORT ON SOURCES AND PATHWAYS OF TMI-2 RELEASES OF RADIOACTIVE MATERIAL. JULY 16, 1979, REVISION 0

The Metropolitan Edison/General Public Utilities Corporation's Preliminary Report discusses pathways for the transport of radioactive material from the reactor building into the other plant building and finally to the environment, isolation of the reactor containment building due to pressure increases, radiation monitoring records, selected portions of the building ventilation systems, and preliminary conclusions.

The report reflects extensive discussion with plant operating personnel. It depends on the reader having the Burns and Roe, Inc., mechanical flow diagrams and general arrangement drawings included in the TMI-2 "Green Book." The report will be a good checkpoint for diagnostics after the plant is accessible because there is considerable detail for each pathway examined.

In Section III, "Pathways for Transport of Radioactive Material Following the TMI-2 Accident," there is a comprehensive compilation of the potential pathways. There were copious details of relief valve locations, setpoint pressures, and discharge recipients. It also included pertinent control room operators' logs and shift foremen's logs and relevant sequences of events.

In the preparation of this report to the President's Commission, information contained in the Met ED/GPU report was checked by this author and found to be accurate.

The appendices were well-developed and rational. The description of the radiation monitors and the responses of the instruments were comprehensive. The development of information to evaluate the potential siphoning of the reactor building sump to the auxiliary was well done. The conclusions appeared to be appropriate up to the point of development.

APPENDIX B

DISCUSSION OF THE NUCLEAR REGULATORY COMMISSION'S EVALUATION OF RADIOACTIVE RELEASE PATHWAYS

The Nuclear Regulatory Commission's evaluation of the release pathways for the TMI-2 accident is written as Details 11 Radiological Aspects, Section IV, subsection on Liquid and Gaseous Pathways. It utilizes the same information as the Met Ed/GPU Preliminary Report on Sources and Pathways of TMI-2 Releases of Radioactive Material.

The NRC discussion is a small part of NUREG-0600, Investigation into the March 28, 1979, Three Mile Island Accident by the Office of Inspection and Enforcement. To understand it adequately, one must first either read and understand the reports (Appendix 1-A, Operational Sequence of Events or Appendix 11-A, Radiological Sequence of Events), be completely knowledgeable of plant designs and operations at Three Mile Island, or be well familiarized from other descriptions of the accident.

One specific item may be somewhat controversial. It was stated that, "Loss of seal water resulted in significant leakage from pumps WDL-P-5 and B, which take suction on the reactor coolant bleed tanks (reference 127)." Loss of seal water should not automatically result in significant leakage although it could if the seals were damaged. The Met Ed/GPU report addresses the problem without conclusions. Final resolution will not be available at least until cleanup of the auxiliary building is accomplished.

In the overall comparison, there is reasonably good agreement about NRC pathways among Met ED/GUP and this author.

APPENDIX C

COMPARISON OF RATING OF PATHWAYS OF RADIOACTIVITY TO THE ENVIRONS

Pathway	Metropolitan Edison/ General Public Utilities Corp.	Nuclear Regulatory Commission	Author of Report to President's Commission
Let-down/Make-Up System	Major pathway to auxiliary bldg	Same	Same
Reactor bldg sump to auxiliary bldg sump	8,4000 gal of low-activity water	Same	Same
Reactor coolant drain tank to reactor coolant bleed tanks	Probably not significant	Neither referenced nor discussed.	Probably added to water inventory in auxiliary bldg.
Reactor coolant drain tank vent to vent gas header	Discussed nominally	Considered significant in damaging vent header system	Considered most important in degrading of vent header components originally
Reactor coolant drain tank vent to reactor coolant bleed holdup tanks	Not of great significance	Same	Same
Reactor coolant pump seal water to seal return coolers	Not discussed	Not discussed	Low probability
Let-down coolers cooling water	Low probability	Not discussed	Low probability
Leakage coolers cooling water	Low probability	Not discussed	Low probability
Make-up pump seal leakage	Low probability	Not discussed	Not discussed; low probability
Relief valve by block orifice valve	Probable path to bleed holdup tanks	Mentioned only	Considered important pathway to bleed holdup tanks
Relief valves on purification demineralizer of let-down/ make-up system	Possible pathway	Considered important pathway to auxiliary building drains	Believe block orifice valve relief valve MU-R1 would open first to prevent discharge from these valves.
Relief valve on make-up tank outlet	Happened	Happened	Happened

REFERENCES

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3. "Preliminary Annotated Sequence of Events, March 28, 1979," Metropolitan Edison/General Public Utilities Corporation, July 16, 1979, Rev. 1.

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30. Control Room Operator and Foreman Logs, TMI-2.

31. Interview with Control Room Operator C (see reference 1).

32. Log typer.

33. Alarm printer.

34. Interview with Radiation/Chemistry Technician D (see reference 1).

35. Make-Up Tank Level Chart Recorder.

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37. Interviews of Shift Supervisors C and E (see reference 1).

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46. SOP 2Z, Metropolitan Edison Co.

47. Drawing Number 2065, Rev. 19, Burns and Roe, Inc.

48. Interview of B. Mehler, Metropolitan Edison Co.

REPORT OF THE TECHNICAL ASSESSMENT TASK FORCE

ON

IODINE FILTER PERFORMANCE

BY

William M. Bland, Jr.

October 1979
Washington, D.C.

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SUMMARY

During the accident at Three Mile Island, a quantity of iodine-131 was detected in the gaseous effluent. This quantity was more than that which would be expected to pass through the filtering system if it performed as designed. Replacement charcoal in the auxiliary building ventilation system and in half of the fuel handling building ventilation system significantly reduced the iodine discharges, suggested that charcoal in the filter trains at the onset of the accident did not perform as expected.

Investigation determined that airflow is designed to normally bypass the filters for control room, auxiliary building, and fuel handling building exhaust, and if the level of radioactivity in the air stream reaches a predetermined level, airflow is diverted to pass through the filters.

Charcoal in use in the filters was purchased in 1975. It met the regulatory requirements in existence at that time but did not conform to the requirements in effect at the time of TMI-2 operating license was issued. The Nuclear Regulatory Commission NRC approved use of the charcoal that was installed and waived the surveillance requirements in the operating license technical specifications for the fuel handling building and control room air-cleaning systems. Such surveillance was intended to verify correct system performance. There was no such surveillance for the auxiliary building ventilation system filter performance.

The air-filtering systems were designed to be used only when needed to remove airborne radioactivity, because of a limited filtering lifetime for charcoal. However, ventilation flow had been through the filters for about one year at the time of the accident. This fact, coupled with the lack of surveillance to verify filter performance, could explain apparently inadequate filter performance during the accident.

Samples of charcoal filters removed from the auxiliary building and fuel handling building filter trains during the accident were tested for removal efficiency. These tests showed a degradation in removal efficiency for methyl iodide the standard test medium. Fuel handling building filter trains (A and B) showed a significant difference in efficiency removal (75.6 percent verus 49.1 percent) suggesting that, in addition to the effect of degradation from one year's operation, one train had been used much more than the others. The auxiliary building filter trains both showed significant degradation.

These efficiency removal figures reflect the effect of the iodine-laden flow during the accident up to the point the filters were taken off line for replacement. Tests are under way to determine the removal efficiency that existed at the start of the accident. Specification charcoal filters were expected to be able to handle any possible iodine effluent effectively.

ANALYSIS

DISCUSSION

As noted in Appendix A, which is a series of daily reports issued by NRC during the TMI-2 accident that began on March 28, 1979, a quantity of iodine-131 was detected in the gaseous effluent from TMI-2. This quantity was more than anticipated to pass through the filtering system that was designed into the plant. It is noted in Appendix A and as shown in a partial summary of Appendices A and B, that when the iodine adsorptive filters -- the activated charcoal -- were replaced in the auxiliary building and in the fuel handling building ventilation systems, the radioactive iodine effluent decreased to expected levels. This indicated that if there were no changes in venting paths or iodine supply, the new charcoal did the expected job, and that the charcoal in the filter trains at the time of the accident had not performed as designed. No allowance has been made for the decay of iodine-131, which has a half-life of 9-/ days; this will show a decrease in activity of the filtered air, also. This apparent poor performance prompted an investigation.

The improvement gained by replacing the charcoal filter elements is described in the April 25, 1979, report in Appendix A, as follows:

> As a result of changing the charcoal filter on the A-Train of the Auxiliary and Fuel Handling Building Ventilation System, the iodine discharges have been reduced by approximately 80%. [Note: Some of this apparent reduction could have been due to decay in activity over the 4-day span of charcoal filter changeout.]

A detailed history of tests and problems with the gaseous radwaste system is presented in reference 1. The most related items in this reference are probably the comments related to the history of the filters prior to the accident.

> The history of these filters prior to the accident may have had a significant impact on their performance during and after the accident. . . .
>
> Since completion of acceptance testing (approximately one year prior to the accident), all ventilation flow from the fuel handling and auxiliary buildings had been through the filter banks (reference 1).

<u>Design.</u> By design, as described in the TMI-2 Final Safety Analysis Report (FSAR) Section 9.4, item f, on page 9.4.1b, reference 2:

> Air flow at the atmosphere cleanup station in the Control Room, and the Auxiliary and Fuel Handling Buildings, normally bypasses the filters. If the level of radioactivity in the air upstream of the filters reaches a predetermined level, the monitoring device will automatically reposition the dampers to reroute flow through the filters. . .

This is confirmed on page 11-7 of the TMI-2 Safety Evaluation Report (SER), reference 3.

Filter History. According to information contained in Appendix C, the charcoal in use in the subject filters at the time of the accident was purchased in 1975 and conformed at that time to the regulatory guide (RG) that was in force at that time (RG 1.52). This charcoal, however, did not meet the requirements of the regulatory guide in force at the time the TMI-2 operating license was issued (RG 1.52, Rev. 1, July 1976. According to references 1 and Appendix C, the NRC, in consideration of the earlier purchase of the charcoal by the utility, issued item F.2 of attachment 2 to operating license DPR-73 (reference 4), which permitted the use of this charcoal by deferring surveillance requirements for the fuel handling building air cleanup (paragraphs 4.9.12.b.2 and 4.9.12.c) (reference 5) and for the control room emergency air cleanup system (paragraphs 4.7.7.1.c.2 and 4.7.7.1.d) (reference 6) specifications which are reproduced below. References 5 and 6 are taken from the TMI-2 technical specification.

Excerpts from Fuel Handling Building Air Cleanup-Surveillance Requirements:

4.9.12.b.2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Postion C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

4.9.12.c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

Excerpts from Control Room Emergency Air Cleanup Surveillance Requirements:

4.7.7.1.c.2 Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b. of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a. of Regulatory Guide 1.52, Revision 1, July 1976.

4.7.7.1.d After every 720 hours of charcoal absorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b. of Regulatory Guide 1.52, Revision 1 July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a. of the Regulatory Guide 1.52, Revision 1, July 1976.

For both cases, the significant one waived is the one that requires laboratory analysis after every 720 hours of charcoal adsorber operation to verify criteria in RG 1.52, Revision 1, July 1976.

It is noted that the technical specification at the time of the TMI-2 Accident did not require surveillance testing of the charcoal filters in the auxiliary building air ventilation system.

EVALUATION

From the foregoing discussion on the facts:

1. although the air filtering systems were designed to be used only when needed by radioactive conditions, the filters had been in continuous use for about a year prior to the initiation of the accident, and

2. through the issuance of item F.2 of attachment 2 to the license, the surveillance requirements were waived that would have verified on a monthly basis the adequacy of the charcoal in the fuel handling building and control room emergency air cleanup system.

These two occurrences could explain why the filter performance was not as expected during the accident and not as good as the performance achieved by the charcoal that was used for replacement during the accident.

The auxiliary building ventilation system charcoal filters had apparently also been used continuously. There was no technical specification requirement for them to undergo surveillance tests. This lack of surveillance testing is borne out by a review of the auxiliary building ventilation system filters Maintenance History that was examined from November 1978 through July 1979. The first sampling for 720 hours surveillance testing is noted on June 12, 1979, Appendix D. Also as noted in Appendix E, a General Public Utilities Service Corporation (GPUSC) man working specifically in this area had not been able to find records, as of Sept. 14, 1979, of any surveillance tests run on these charcoal filters prior to the accident.

It is noted that removal efficiency tests conducted by Nucon* on the samples of the charcoal filters that were removed from the auxiliary building and fuel handling building filter trains during the March 28, 1979, accident, Appendices E and F, show degradation in removal efficiency of methyl iodide. The B filter train in the fuel handling building system shows significant degradation, 49.1 percent compared to 75.6 percent on train A. This indicates that possibly, in addition to having been subjected to flow for almost a year, that the fuel handling building duct system may have had a more unbalanced flow through the filter trains, or much more exposure to Iodine in B train, than the duct system in the auxiliary building, where the filter trains showed more comparable degradation, as shown below.

Nuclear Consulting Services, Inc., Columbus, Ohio, performs the testing for Metropolitan Edison Company.

The results of the removal efficiency test conducted to date on the charcoal removed from TMI-2 during the accidnet (Appendices E and F) are as follows

Location	Removal Efficiency @ 95 percent Relative Humidity
Auxiliary Building, A-Train	69.5 percent
Auxiliary Building, B-Train	56.0 percent
Fuel Handling Building A-Train	75.6 percent
Fuel Handling Building B-Train	49.1 percent

New charcoal, by reference 7, should have a methyl iodine removal efficiency of 99 percent. The removal efficiency requirement has been increased in subsequent issues of the Regulatory Guide.

Replacement of the charcoal during the accident was done in the following sequence (Appendix B).

Auxiliary Building, A Train	April 20, 1979
Fuel Handling Building, A Train	April 24, 1979
Auxiliary Building, B Train	April 25, 1979
Fuel Handling Building, B Train	Approximately May 23-24, 1979

Change out of the charcoal in the fuel handling building B train, which has the lowest tested removal efficiency of the samples removed during the accident, as noted above, was accomplished last. The reason for the changeout sequence according to a telephone conversation with Mr. Montgomery of GPUSC on Sept. 28 was that when change out of the charcoal filter in the B train was first attempted, on about April 20, it was found to be radioactively too "hot" to handle. He noted that a radiation level of at least one em/hour was measured on April 21. Changeout of train B is reported to have been accomplished about May 23-24, after sufficient decay had occurred. These dates and the radiation level are subject to confirmation. Radiation level readings made on the other three charcoal trains at changeout of the charcoal during the accident have been requested. Response to date reports that the radiation level on the charcoal in the fuel handling building A train ranged 150-350 millirems/hour when charcoal change out was begun on April 21.

It is understood that further testing is underway by Nucon at this time (Oct. 12, 1979), may better define the removal efficiency of these charcoal filter trains at the time of the beginning of the TMI-2, March 28, 1979, accident. This additional information may give more understanding as to the reason why the filters in place at the time of the accident performed so poorly.

FINDING

It is probable that the ventilation flow through the filter trains in the auxiliary building and the fuel handling building ventilation systems in the year prior to the accident significantly decreased the removal efficiency of the charcoal filtering elements. The waiver, granted by NRC, of the periodic surveillance testing requirement for the fuel handling building charcoal filters the omission of a periodic surveillance requirement for the auxiliary building charcoal filter, and the use of charcoal that did not meet the minimum requirements at the time of TMI-2 licensing contributed to the use of charcoal filter elements that did not accomplish the required filtering at the beginning of the accident at TMI-2.

REFERENCES

1. NUREG 0600, "Investigation Into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement," August 1979. Accession #9290012.

2. TMI-2 Final Safety Analysis Report (FSAR), Section 9.4. Accession #920002.

3. TMI-2 Safety Evaluation Report (SER), NUREG 0107. Accession #6290170.

4. Item F.2 of attachment 2 to operating license OPR-73. Accession #9290004.

5. TMI-2 technical specification surveillance requirements, 4.9.12. Accession #9290005.

6. TMI-2 technical specification surveillance requirements, 4.7.7.1. Accession #9290006.

7. Regulatory Guide 1.52, July 1976: Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety, Featuring Atmosphere Cleanup System Air Filtration and Absorption Units of Light-water-Cooled Nuclear Power Plants. Accession #9290007.

APPENDICES*

A. "Preliminary Notification of Event on Unusual Occurrences," issued relative to the TMI-2, March 28, 1979, accident. Accession #9290000.

B. Summary of TMI-2 iodine filter activity from reference 1. Accession #9290001.

C. Memorandum for Record, "Status Report on Filter Elements from TMI-2," William Bland, Sept. 20, 1979. Accession #9290003.

D. Memorandum for record, "Auxiliary Building Exhaust Filter Maineenance History 10/1/79 - 7/31/79," Art Carr, Sept. 10, 1979. Accession #9290008.

E. Memorandum for Len Jaffe, "Update on Iodine Filter Data," William Bland, Oct. 11, 1979. Accession #9290009.

F. Letter, Shaw, Pittman, Potts, and Trowbridge, **transmitting "Analysis** of the Adsorbers and Adsorbents from Three Mile Island Unit No. 2," June 11, 1979. Accession #9290011.

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* These documents are part of the Commission's permanent records and will be available in the National Archives.

REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

RECOVERY:
TMI-2 CLEANUP AND DECONTAMINATION

BY

Bruce Mann

October, 1979
Washington, D.C.

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I. FINDINGS

1. The TMI-2 facility cleanup and decontamination represent a task which is greater in magnitude and complexity then previously encountered in the U.S. commercial nuclear power industry.

2. Cleanup cost is expected to be between $100 and $200 million. This does not include costs for refurbishment and return to service.

3. Overall planning and task definition, and the development of a preliminary schedule, have been completed. The entire cleanup is expected to take at least 2 years.

4. In the opinion of knowledgeable experts, the practical and technical experience base within U.S. governmental and civilian organizations is adequate to perform the cleanup. The expertise of the U.S. Department of Energy (DOE) contractors is being utilized to assist the utility and its industrial contractors. Engineering and chemical process development work will be required for several of the tasks.

5. Continued presence of the radioactive materials presently dispersed in the large volumes of air and water contained in the facility present a risk of uncontrolled release to the environs. Completion of the cleanup and decontamination will result in the reduction of exposure risks to both workers and members of the public.

6. The cleanup will produce large volumes of radioactive waste materials (over 500,000 cubic feet) that must be disposed of. Final disposition of the radioactive waste and the damaged reactor core is yet to **be determined.**

II. INTRODUCTION

As a result of the accident on March 28, 1979, the TMI-2 facility became extensively contaminated by radioactive fission products released from the damaged reactor fuel. The status of the facility and the extent of contamination shortly after the accident is described in the Oak Ridge National Laboratory (ORNL) report (reference 6).

At present, the radioactive material remaining in the facility includes: the damaged core itself; fuel debris, which possibly has been transported to locations in the primary coolant system; fission products dissolved and suspended in the primary coolant and in water contained in the reactor containment building and the TMI-2 auxiliary building; gaseous radioactivity in the containment building atmosphere; and radioactively contaminated materials in various forms which have precipitated and settled onto numerous surfaces (equipment and building interiors) in the TMI-2 containment, auxiliary, fuel handling, and diesel generator buildings (reference 6).

The bulk of the remaining radioactive material that is distributed outside of the fuel is contained in several volumes of water. This water contains in total approximately 850,000 curies of long-lived fission products (mostly cesium-137 and strontium-89 & 90) and consists of approximately: 90,000 gallons in the primary coolant system, 600,000 gallons in the reactor containment building, and about 380,000 gallons in several large tanks located in the TMI-2 auxiliary and fuel handling buildings. The atmosphere in the containment building contains about 51,000 curies of krypton-85 (half-life 10.7 years, a noble gas) (reference 34).

Floors, sumps, and equipment surfaces in the above mentioned facilities were extensively contaminated, largely due to flooding and subsequent water leakage from tanks onto floors and sumps. No estimate is available regarding the total amount of radioactive material that is involved in this contamination, nor of the total number of curies remaining, but it is generally comprised of the same isotopes as contained in the inventory of contaminated water in the TMI-2 facility (reference 34).

III. OVERVIEW

In general terms, the cleanup refers to the concentration and removal of radioactive materials from the various locations in the facility where they were distributed as a result of the accident. In this discussion, it includes the removal of the damaged reactor core and decontamination of the primary coolant system. It does not include repair or replacement of damaged equipment or activities related to refurbishment and return to service. The cleanup and decontamination must be conducted regardless of the final disposition of the TMI-2 facility (reference 3).

The principal tasks comprising the cleanup include: contaminated water treatment,. facility and equipment decontamination, and radioactive waste handling and disposal. In terms of major TMI-2 facilities, the auxiliary, fuel handling, diesel generator, and reactor containment buildings are involved.

For purposes of radioactive water management and decontamination, three levels of contaminated water have been designated. Low-activity water is that containing less than one microcurie per milliliter; intermediate level, between one and 100 microcuries per milliliter; and high level, greater than 100 microcuries per milliliter. These definitions were originally developed in terms of iodine-131, which immediately following the accident was the most significant constituent from a hazard protection perspective (reference 6). At the present time, less than one curie of 1-131 (half-life, 8 days) remains of the original inventory. The present water level definitions are based on cesium-137 concentrations (reference 34).

A system to process (decontaminate) the 380,000 gallons of intermediate-level water contained in the TMI-2 auxiliary and fuel handling building tanks has been designed and installed on the site (reference 6). This system, known as EPICOR II (after the company that developed the system), was designed to process large volumes of water containing between one and 100 microcuries per milliliter of cesium and iodine (reference 21).

Work is under way on a system to process the high activity water in the primary system and the reactor containment building. It is being designed to process water containing greater than 100 microcuries per milliliter of cesium and strontium. A summary of the radioactive species and their concentrations in the intermediate- and high-level water inventories is contained in the ORNL report (reference 6).

Both of these systems utilize filtration and ion-exchange to concentrate the radioactivity. They are designed for remote operation with shielding and effluent control equipment, because of the large amounts of radioactivity involved (reference 6). The spent resin and sorbents will require special handling and shielding due to the high radiation fields from the concentrated cesium-137. Because of radiation damage limitations, the high activity water processing system will use inorganic ion-exchange resins (zeolites). Details of processing system designs are contained in the ORNL report (reference 6).

The NRC has required that resins from the EPICOR II system be solidified on-site prior to shipment, and it is expected that the resins from the high-level water treatment system also will be solidified (references 22, 23). Additional discussion of radioactive waste handling and disposal is contained in section V.

Removal of radioactivity from equipment and building surfaces is largely a hands-on operation. It consists of wet and dry vacuuming, mopping, and wiping (reference 6). Semi-portable equipment, such as degreasing units, ultrasonic cleaners, and electropolishing machines, also are used to decontaminate small equipment items (reference 26). Decontamination involves work in high radiation areas and requires protective clothing, and in some cases, the use of respiratory protection from airborne radioactivity. In April, work was started on decontamination of the TMI-2 diesel generator building, and in May, on the auxiliary and fuel handling buildings. As of the end of September, the decontamination of these building floor areas was about 90 percent complete (reference 26).

By far, the largest task in the cleanup is the containment building decontamination. The Bechtel Corporation was hired by General Public Utilities Corporation (GPU) to assess the overall job and to develop preliminary task scopes, schedules, and costs. The planning study was completed in July (reference 4). The study identified the basic tasks as: containment atmosphere purging, containment building sump water treatment, containment reentry and decontamination, reactor coolant processing, removal of the reactor core, and decontamination of the primary system (reference 4). A summary of the containment building cleanup tasks is provided in the ORNL report (reference 6).

A preliminary cost and schedule assessment by Bechtel estimates that the containment building cleanup will take approximately 2 years after initial entry (reference 5). The study does not specifically estimate the cost of cleanup, but includes an estimate of the overall recovery and return to service. However, if costs are prorated by subtask, the total for the containment building cleanup (including support services and a 33 percent contingency) is $200 million. A study of accident costs for the President's Commission estimated facility decontamination and fuel removal costs of $90 to $130 million (reference 32).

The Bechtel study identified a number of contingencies that could affect schedule and cost estimates. These include more severe radiation or structural damage conditions than anticipated, labor shortages, regulatory and licensing delays, and legal or political problems (reference 5).

IV. EXPERIENCE AND PROGRESS TO DATE

Low-activity water processing has been under way at TMI since early April following the accident. A portable treatment system (EPICOR I) was brought in to process low-activity water held in tanks in the TMI-1 facility. Some of this was pre-accident water from both TMI-1 and 2, mixed with a relatively small amount of post-accident water from TMI-2. As of mid-August, about 75,000 gallons of this water had been decontaminated by EPICOR I and released to the Susquehanna River in accord with NRC regulations (reference 34). In-leakage to this water in TMI-1 is accruing at the rate of about 300 gallons per day. Water processing by EPICOR I continues at the rate of about 2,000 gallons every 5 days (reference 17). At the end of September, the inventory of low-activity water in TMI-1 was about 115,000 gallons (reference 34).

An additional quantity of low-activity contaminated water was created during the accident by leakage through the TMI-2 B steam generator into the TMI-2 secondary water system. The secondary water has since been processed by the TMI-2 condensate polishing system. At present, the B steam generator contains cesium-137 concentration of about 0.03 microcuries per milliliter (reference 15). Processing of water by the EPICOR I and TMI-2 condensate demineralizer systems has resulted in the production of spent resins that must be disposed of. These are discussed in more detail in the following section.

In-leakage in TMI-2 continues to contribute to water management problems at TMI. Leakage into the intermediate-level water storage volumes is occurring at the rate of about 800 to 1,000 gallons per day (reference 3). This occurs primarily due to leakage from river water service system pumps in the TMI-2 auxiliary building. The water drains to the contaminated sumps in the TMI-2 auxiliary building, hence it must be collected and added to the contaminated water storage (reference 17). At this rate of in-leakage, Met Ed projected that by the end of October, they would have filled all secure contaminated water storage capacity at TMI-2 (reference 3). The recent NRC decision to permit use of EPICOR II for processing intermediate-level water should provide additional storage margins (reference 22). Court actions have been filed against the utility and the NRC, which seek to prohibit the release of water processed by EPICOR II to the Susquehanna River (reference 21). The utility has prepared an assessment of alternatives for disposing of the water. The NRC plans to perform an environmental assessment of the proposed alternatives. The disposition of the treated water will be determined following the completion of the NRC review. In the meantime the treated water must be retained on the site (reference 22).

In-leakage to the approximately 600,000 gallons of high-level activity water in the containment building continues at the rate of about 500 to 900 gallons per day (references 23, 17). This leakage, apparently is due primarily to pump seal leaks inside the containment building and cannot be reduced until new primary coolant pressure and volume control, and heat removal systems are operable. It has been estimated that about 100,000 gallons of additional water in the containment building will raise the water level to the point where it affects the maintenance of the reactor in a stable cooling mode (reference 23). At the current rate of in-leakage, this point will be reached in about 100 to 120 days (from the time of this writing). However, the new systems designed and installed since the accident to maintain the plant in a stable long-term cooling mode are expected to be in operation before that time (reference 34). This should eliminate potential problems from the slowly rising water level in the containment building over the next several months timeframe.

The system for processing the containment building water is not exepcted to be operational until August 1980, and it is possible that water transfers from the building may be necessary before then (reference 35). Processing of intermediate-level water by EPICORE II in the interim should make additional tankage available in the TMI-2 fuel handling and auxiliary buildings if it becomes necessary to transfer water from the containment building.

With the assistance of several contractors, Met Ed/GPU has examined alternatives for the disposition of the contaminated air in the containment building (reference 34). They have prepared a proposal to release the air (after filtration) slowly to the outside atmosphere under controlled conditions over a 2-month period (reference 34). The release of krypton-85 under these conditions is not expected to result in exposure to any off-site individuals in excess of applicable NRC regulations for routine operation of a nuclear power plant (reference 34). The 51,000 curies of krypton in question is within the range of the annual quantity of radioactive noble gas released from an operating light-water reactor in the United States (reference 24). The proposal to perform the containment air purging and the associated safety and environmental analyses are under review by the NRC (reference 20).

In response to the accident, extensive environmental radiation surveillance activities in the area surrounding Three Mile Island were conducted by the state of Pennsylvania and several federal agencies. The various agency efforts have since been consolidated into a comprehensive long-term surveillance program. The program is designed to provide environmental monitoring for cleanup and recovery operations, and contains provisions for rapid assessment of potential uncontrolled releases from the facility (reference 11). In addition, a protocol has been established between Met Ed, NRC, and the state of Pennsylvania for notification and monitoring of all radioactive waste shipments leaving the site (reference 34).

At present, the containment building remains sealed to contain the contaminated air and water. The building atmosphere is maintained at a slightly negative pressure to further reduce the likelihood of leakage to the outside (reference 36). A penetration was made into the containment building in September by drilling a 2-inch hole through a blanked off access port. This allowed the collection of containment water and sediment samples. Analysis of the inside surface of the drilled plug also was performed to assess the radioactivity plate out on the inside of the containment building (reference 6). Work is currently under way to drill a larger opening into the containment building, which will allow insertion of optical devices and radiation measuring equipment for further assessment of conditions inside (reference 7). Human entry is not planned until the containment building air has been purged. Development of detailed decontamination procedures and plans cannot be made until entry can be accomplished and direct assessment of conditions inside completed. The earliest that human entry is expected is in late January 1980 (reference 34). Actual decontamination of the reactor building is not planned to begin until the contaminated water is removed from the building. Present planning estimates indicate that decontamination work will begin in October 1980 (reference 34).

As mentioned previously, decontamination and cleanup entail work in radiation fields, and workers are exposed to the possibility of inhalation and ingestion of radioactive materials. Most of the actual hands-on work performed thus far (auxiliary, fuel handling, and diesel generator buildings) has been performed by volunteers from within the Met Ed organization (reference 26). Project management, technical support, and radiation monitoring services are provided by several contractors. The work requires extensive health physics support for radiation exposure and contamination control (reference 26).

Some information is available regarding the radiation exposure experience associated with cleanup efforts conducted thus far. Preliminary data for the third calendar quarter of 1979 (July-September) shows the collective exposure for decontamination workers to be about 26 person-rems. A total of 182 workers was involved (reference 29). By way of comparison, the 3-month total for June through August for all on-site personnel was 285 person-rems. The average on-site population during this period was about 3,000 (reference 30). Thus far, whole-body counting and bioassay results on decontamination workers have not shown detectable uptake of radionuclides in any individuals (reference 30).

The continued presence of the contaminated water and contaminated areas in the facility provides radiation exposure to the personnel whose presence in radiation areas is required for maintainence and facility support operations. In late August, five workers performing maintenence work on a contaminated water storage system in the TMI-2 auxiliary building received radiation exposures in excess of NRC limits. They received exposures to the skin and extremeties in excess of NRC quarterly limits due to unexpectedly high beta radiation fields that were not adequately understood by radiation monitoring personnel (reference 13).

The NRC staff has expressed concerns regarding the adequacy of the health physics (radiation protection) program at TMI. The concern has been that the Met Ed/GPU health physics organization lacked adequate professional staff and was not effectively represented in the on-site management structure (reference 25). The utility has taken steps to strengthen the health physics capability by the hiring of additional professional personnel (reference 30). The NRC has recently formed a special advisory panel for the purpose of reviewing the health physics program at TMI and providing recommendations for improvements, should additional improvements be warranted (reference 18).

V. RADIOACTIVE WASTE DISPOSAL

Large quantities of radioactive solid materials will be produced as a result of the cleanup and decontamination. At present, only rough estimates are available of quantities expected. The Bechtel study estimated that somewhere between 2,000 and 3,000 shipments (truckloads) will be produced in the overall facility cleanup (reference 4).

A variety of types of waste materials will be produced during the cleanup. Each requires somewhat different handling and packaging. All solid wastes will eventually be shipped off the TMI site for disposal.

As discussed in section III, processing of the large volumes of contaminated water produced by the accident will yield resins, filter media, and sorbent materials containing the concentrated radioactivity from the water. Resins and sorbents from low-activity water processing, i.e., the EPICOR I resins and the accumulated resins from in-plant (pre-accident) liquid waste treatment systems are estimated to contain from 0.01 curies to about 3 curies per cubic foot of cesium-137 (reference 9). The total volume of resins from low-activity water processing is estimated to be about 3,000 cubic feet. If solidification of these resins is required, the volume of material to be disposed of will be approximately doubled (reference 9).

Resins and sorbents from processing intermediate- and high-activity water will contain from about one to several thousand curies of cesium and strontium per cubic foot (reference 9). A preliminary estimate of the volume of these materials indicates that about 6,000 cubic feet of this material will be produced (reference 9). This material will be solidified on-site prior to shipment, as indicated in section III. The volume estimate is for the resins in solidified form. The estimates include the volume of solidified resin plus permanent containers. Temporary shielding and transportation casks or containers are not included. Details of resin characteristics, handling, and shipping are given in the ORNL report (reference 6).

In terms of volume, a major source of wastes will be what is generally referred to as dry compactible materials. These materials include expendable items, such as protective clothing worn by workers, rags, smears, small tools, and miscellaneous materials used in and generated by decontamination work. The bulk of this material will be compacted on-site into industrial-grade, 55-gallon drums (reference 6). This material, in general, will be of relatively low radioactivity content and will not require special shielding.

Noncompactible dry solid materials will also be generated. These materials include damaged equipment removed from the facility, temporary shielding and construction materials used in the cleanup, contaminated expended charcoal beds and filters from air-cleaning systems, and a variety of miscellaneous items (reference 4, 6). Those materials of appropriate size that do not present special radiation shielding requirements will be packed and shipped in wooden cartons reference 4). This and the compacted material together are estimated to amount to about 400,000 cubic feet for the entire cleanup operation (reference 4).

It is expected that most of this material will be of low-activity content and will be suitable for disposal at a commercial radioactive waste burial site.

The Bechtel study recognizes that the cleanup may produce materials that will require special handling. This category of material includes large damaged components such as main coolant pumps and containment building air coolers, which either due to their size, associated high radiation levels, or both, will involve extra shielding or oversize (and perhaps overload) shipments. An estimate of 20 special shipments for this category of material has been made (reference 4).

Decontamination of the reactor containment building and the equipment inside will require the use of large quantities of water and decontamination solutions. Bechtel has estimated that somewhere between 3 and 9 million gallons will be required in all (reference 4). The processing of this water will yield a significant amount of concentrates that must be solidified. Due to the complex chemistry of these decontamination solutions, it is planned to treat them using a large evaporator facility that is being designed for installation on the TMI site (reference36). This will yield a concentrate-sludge or evaporator "bottom" containing the radioactive materials that will then be solidified for disposal. It is planned to reuse decontamination solutions processed by the evaporation to reduce the total amount of water that must be handled (reference 31). Approximately 15,000 cubic feet of solidified wastes are expected from evaporator concentrates (reference 4).

Special handling procedures will be required for the damaged fuel and reactor core components due to the intense radiation levels associated with this material (references 4, 6). Preliminary examination of the requirements for the removal of this material from the reactor vessel have been made by Bechtel (reference 4). A more detailed analysis of the technical engineering requirements for handling, shipping, and for the ultimate disposition of the fuel is now under way (reference 33).

The analysis also includes a survey of various civilian and government-owned facilities that could be used to prepare the fuel for ultimate disposition.

There are several uncertainties regarding the ultimate disposition of the radioactive wastes from the TMI cleanup. Most of the wastes, as measured by volume, will be of low specific activity (LSA). The Bechtel study defines this material as containing less than one curie per drum or container (reference 4). Somewhere between 400 and 500 thousand cubic feet of LSA material are expected altogether. The Bechtel study also identifies a category of solid packaged wastes as intermediate-level wastes. This category is defined as containing between one and 10 curies per container with external radiation dose rate of one Rad per hour or less measured one foot from the container surface. Approximately 20,000 cubic feet of this material have been projected (reference 4). The low- and intermediate-level wastes expected from TMI are similiar to material that is routinely disposed of at commercial burial sites.

Following the accident, Met Ed shipped some LSA waste in accord with previous practice to the commercial burial site at Barnwell, S.C. However, the governor of South Carolina intervened, and post-accident wastes from TMI were prohibited from the Barnwell site (reference 27). Subsequent to that incident, arrangements have been made for the receipt of about 200 shipments of wastes from TMI by the commercial disposal site at Richland, Wash. (reference 35). This agreement includes wastes from the fuel handling and auxiliary building decontamination and resins from the EPICOR I and II systems. These shipments are expected over the next 12 months to 2 years. As of mid-September 1979, 12 shipments of post-accident waste have gone to the Richland site (reference 19).

Recent events have created new uncertainties regarding the availability of disposal sites for TMI wastes. Concern with site management practices and with waste transportation problems has led to the recent closing of the commercial burial sites at Richland, Wash., and Beatty, Nev. (reference 15). At present it is not known how long these sites will remain closed.

The NRC has been concerned with the viability of the commercial radioactive waste burial sites. In 1975, there were six operating sites in the country; at present, only the Barnwell, S.C., site remains open, and it is not available for TMI wastes. In view of the uncertain viability of several of the commercial burial sites, the NRC had previously requested that DOE prepare a contingency plan for the use of DOE-owned shallow land burial sites for receipt of commercial radioactive wastes (reference 15). In view of the recent site closings in Nevada and Washington, the NRC is currently reviewing alternatives for both interim- and long-term waste disposal options. At this writing, it is not known what solutions will be proposed.

At present, the ultimate disposition of the high activity resins is not known. With concentrations of cesium and strontium of up to several thousand curies per cubic foot, this material is much higher in activity concentration than is generally disposed of in shallow land burial facilities. Waste classification regulations currently under consideration by NRC would preclude this material from disposal at existing shallow land burial facilities (references 15, 28).

VI. DISCUSSION AND CONCLUSIONS

A number of preliminary conclusions regarding the cleanup can be drawn. It is clear that the cleanup of the TMI-2 facility from the accident of March 28, 1979, represents a task in both magnitude and complexity that has not been previously encountered by the U.S. civilian nuclear power industry. This is easily borne out on the basis of preliminary cost estimates for the cleanup, which range from about $100 to $200 million. It is also apparent that extensive experience in the decontamination and recovery of a large number of nuclear facilities has been gained over the past 30 years by both governmental and civilian organizations. Successful completion of cleanup and recovery operations that include the type of tasks faced by the TMI-2 cleanup have been performed at various facilities, including the handling of damaged irradiated reactor cores (references 1, 10).

In the opinion of knowledgeable experts, the practical and technical experience base within U.S. governmental and civilian organizations is adequate to perform the cleanup, (references 8, 14). Engineering and chemical process development work is required, however, and is under way for various tasks. It is likely that facilities and expertise of DOE contractors will be necessary for the removal, handling, and disposition of the damaged reactor core. This depends in part on decisions yet to be made regarding the interim and ultimate disposition of the fuel material after it is removed from TMI.

Additional engineering development work may be required in order to satisfy environmental release constraints that could be applied to the TMI-2 cleanup. For example, if Met Ed is precluded from disposing of the 51,000 curies of krypton-85 presently in the containment building air, cryogenic trapping, adsorption on charcoal, or concentration and storage under pressure will have to be considered. None of these potential alternatives have been demonstrated successfully on the scales necessary for TMI-2 (reference 12).

Continued presence of materials in the TMI facility dispersed in the large volumes of air and water present increased risk of uncontrolled release to the environs. The orderly, systematic cleanup and decontamination of the facility with concentration and confinement of the radioactive materials would result in an overall reduction in exposure risk to both workers and members of the public living in the vicinity of TMI.

METHODOLOGY

During the course of the investigation by the President's Commission, it became apparent that a major aftermath of the accident was the task of cleaning up the extensive radioactive contamination in the facility. The Commission staff requested Oak Ridge National Laboratory (ORNL) to provide a description of major decontamination and cleanup tasks at TMI. This has been provided in the form of a technical report to the Commission (reference 6).

The present report has been prepared to provide a summary with less technical detail and to cover additional topics of interest not included in the ORNL report. In addition, this report provides more recent information on the status of several cleanup tasks. In the collection of information for this report, the author relied heavily upon discussions with representatives and employees of Metropolitan Edison Company (Met Ed), General Public Utilities Corporation (GPU), Oak Ridge National Laboratory, South Carolina Energy Research Institute, the U.S. Nuclear Regulatory Commission (NRC), the U.S. Department of Energy (DOE), Bechtel Corporation, and the Electric Power Research Institute (EPRI).

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