Staff Reports To
The President's Commission On

THE ACCIDENT AT THREE MILE ISLAND

Reports Of The Technical Assessment Task Force, Vol. III

THE PRESIDENT'S COMMISSION ON
THE ACCIDENT AT
THREE MILE ISLAND

JOHN G. KEMENY, CHAIRMAN
President, Dartmouth College

BRUCE BABBITT
Governor of Arizona

PATRICK E. HAGGERTY
Honorary Chairman and
General Director
Texas Instruments Incorporated

CAROLYN LEWIS
Associate Professor
Graduate School of Journalism
Columbia University

PAUL A. MARKS
Vice President for Health Sciences
and Frode Jensen Professor
Columbia University

CORA B. MARRETT
Professor of Sociology and
Afro-American Studies
University of Wisconsin-Madison

LLOYD McBRIDE
President
United Steelworkers of America

HARRY C. McPHERSON
Partner
Verner, Liipfert,
Bernhard, and McPherson

RUSSELL W. PETERSON
President
National Audubon Society

THOMAS H. PIGFORD
Professor and Chairman
Department of Nuclear
Engineering
University of California
at Berkeley

THEODORE B. TAYLOR
Visiting Lecturer
Department of Mechanical
and Aerospace Engineering
Princeton University

ANNE D. TRUNK
Resident
Middletown, Pennsylvania

Stanley M. Gorinson
Chief Counsel

Vincent L. Johnson
Director of Technical Staff

Barbara Jorgenson
Public Information Director

REPORTS OF THE TECHNICAL ASSESSMENT TASK FORCE

ON

SELECTION, TRAINING, QUALIFICATION, AND
LICENSING OF THREE MILE ISLAND
REACTOR OPERATING PERSONNEL

TECHNICAL ASSESSMENT OF OPERATING,
ABNORMAL, AND EMERGENCY PROCEDURES

CONTROL ROOM DESIGN AND PERFORMANCE

VOLUME III

October 1979
Washington, D.C.

This document is solely the work of the Commission staff and does not necessarily represent the views of the President's Commission or any member of the Commission.

For sale by the Superintendent of Documents, U.S. Government Printing Offke
Washington, D.C. 20402

Stock Number 062-003-00730-4

TECHNICAL ASSESSMENT TASK FORCE

Leonard Jaffe, Head
William M. Bland, Jr.
Robert English
Ronald M. Eytchison
Dwight Reilly
William Stratton
Jasper L. Tew

CONSULTANTS

Louis Baker
Robert D. Burns, III
Arthur M. Carr
Paul Cohen
Michael Corridini
Hans Fauske
Herb E. Feinroth
Mario Fontana
Frederick Forscher
Peter Griffith
Jerry Griffiths
Raymond Heiskala
George Inskeep
John Ireland
Walter Kirchner
Dave Latham
Harry Lawroski
Saul Levy
John Mailer
Peter Mailer
Peter Mast
Margaret Mlynczak
Frank Muller
John Orndoff
Matthew Opeka
Dana Powers
A. B. Reynolds
Warren Rosenau
Robert Seale
Michael Stern
Neil E. Todreas
Beverly Washburn

TABLE OF CONTENTS

REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

SELECTION, TRAINING, QUALIFICATION,
AND LICENSING OF THREE MILE
ISLAND REACTOR OPERATING PERSONNEL

BY

Ronald M. Eytchison

October 1979
Washington, D.C.

TABLE OF CONTENTS

I. SUMMARY

As a part of the effort to identify and evaluate the possible causes of the Three Mile Island accident, an analysis of operator training, qualification, licensing, selection, and manning was conducted by the staff. The study included review of documents, interviews, and depositions at Three Mile Island, Babcock & Wilcox, and the Nuclear Regulatory Commission (NRC) during June, July, and August 1979. Analysis of the information obtained was conducted almost exclusively by the writer.

This paper examines the roles of the actors involved in training and it reviews the various programs which were intended to staff Three Mile Island with sufficient numbers of competent, trained operators and supervisors. The analysis includes a review of the regulations concerning operator training and licensing; describes how the requirements were implemented by the NRC, Metropolitan Edison Company (Met Ed), and Babcock & Wilcox Company (B&W), and then evaluates the programs conducted by these three organizations.

Perhaps the most significant feature of the regulations is that there is no regulation regarding operator selection and training; the NRC has no minimum eligibility standards for the qualification of operators. Rather, the NRC endorses a standard established by the American Nuclear Society (ANS) pertaining to the selection and training of nuclear power plant personnel. This standard (ANSI 18.1-1971) includes <u>recommendations</u> to the utility concerning selection, training, and qualifications. Reactor operator candidates do not have to meet any <u>requirements</u> concerning minimum education, experience, reliability, criminal record, or stress fitness.

With regard to reactor operator licensing, the regulations require that licensing examinations include questions on construction, design features, operating characteristics, and procedures. There is a lack of emphasis on the comprehensive knowledge of theory, principles of operation, kinetics, thermodynamics, and so on, which would enable operators to correctly interpret information available to them in the control room. Review of typical examination contents indicates the examinations are consistent with the regulations; they do not ensure that license candidates have an in-depth knowledge of nuclear reactor theory, design, and operation.

The NRC's involvement in operator training was perhaps limited by a shortage of resources in the Operator Licensing Branch (OLB). This group was understaffed and had to utilize examiners without reactor operating experience. Overall, the NRC did not cause a sufficiently comprehensive understanding of reactor plant design and operation by those persons who are licensed to operate the reactors or supervise their operation.

The Babcock & Wilcox Company (B&W) had a crucial role in the training of Three Mile Island reactor operators. Because Met Ed did not have its own simulator, B&W, under contract, provided the only practical training given to Three Mile Island operators in operating and emergency procedures. Despite the importance of B&W training, neither the company management, plant designers, nor other engineers had significant influence on the content of training courses or the manner in which they were conducted. Few engineers at B&W, in fact, had first-hand experience in nuclear reactor operations.

Review of the curricula at Babcock & Wilcox indicates that, in general, the simulator training courses lacked depth. Evolutions and drills in the simulator were relatively simple. Casualties involved single failures and were carried only through the immediate action rather than to their logical conclusions. The trainees did not function as a crew and no one was in overall charge.

Operators at the Davis-Besse-1 plant experienced a transient on Sept. 24, 1977, similar to that which later confronted operators at Three Mile Island. They interpreted parameters and took actions similar to their Met Ed counterparts. Significantly, the Davis-Besse operators realized that they were mistaken in sufficient time to avoid core damage.

The lessons of Davis-Besse and the potential hazards of that transient were recognized by Babcock & Wilcox Nuclear Power Generating Division (NPGD) managers who were in a position to convey these lessons to utilities and incorporate the lessons into the B&W simulator training program. This was not done before the TMI accident.

Review of training conducted at B&W indicates that the manner in which the plant would be expected to respond to a small-break loss-of-coolant accident from the steam space of the pressurizer was not included in operator training. In fact, the simulator was unable to reproduce a transient of the nature of that which occurred at TMI-2 because it could not simulate voiding in the coolant system outside the pressurizer. Students were not taught about saturation conditions in the reactor coolant system. The principle of keeping the core covered was not specifically treated in the training program. Babcock & Wilcox did not instruct trainees one way or the other about allowing the pressurizer to go solid when the reactor is shut down.

The success rate of TMI operators on NRC licensing examinations has been better than average, suggesting an above-average training program. However, training received by operators at TMI did not prepare them sufficiently to cope with the March 28 transient. Underlying this was insufficient training department staffing and the lack of management involvement in operator training. Members of the training department were no more qualified educationally than the trainees.

The replacement operator program which prepared reactor operator candidates for licensing was done essentially on a self-study basis. Although 9 months were devoted to this program, little emphasis was given to theory, application of theoretical to the practical, or the principles of operating and casualty procedures. Required study did not cover thermodynamics and such concepts as saturation, enthalpy, decay heat production, or solid system operation. Rather, emphasis was on systems, equipment, and procedures. Operator trainees were not provided with a fundamental, comprehensive understanding of their reactor plant design and operation which would enable them to recognize the significance of a set of circumstances not explicitly predicted by the operating procedures and which would lead them to place the plant in a safe condition.

Similar weaknesses were noted in the operator requalification program. Material was not treated in depth. Lectures covered did not fulfill the requirements of the Met Ed administrative procedure on requalification. An average of only about one hour per week was devoted to lectures on topics related to reactor operation. Again, there was a lack of instruction which would enable operators to understand the significance of reactor plant phenomena.

With respect to training specifically related to the accident, there is no evidence that operators were taught about pressurizer level versus reactor coolant system (RCS) pressure. Instruction did not adequately cover recognition of a loss-of-coolant accident, saturation conditions, the need to remove decay heat, or the significance of high temperatures and pressures in the reactor building.

Fundamental problems related to the training and licensing of Three Mile Island operators might be highlighted as follows:

- There was a gulf between the operators or operationally oriented personnel and the managers or other decision makers in the NRC, at Met Ed, and at Babcock & Wilcox. Few communications took place between Babcock & Wilcox management/engineering and simulator instructors. The NRC Operator Licensing Branch, which set and enforced the standards for operator training, was understaffed and lacked outside direction. Three Mile Island management did not consider itself responsible for operator training.

- The training standard was low and did not require that the operators be provided with the analytical tools necessary to operate a nuclear reactor.

- There was no effective mechanism for learning from the mistakes of others. The system was such that, in large part, the utilities had to learn for themselves. Quite probably, the accident at Three Mile Island would not have occurred if the operators had been thoroughly and comprehensively trained on the lessons of Davis-Besse.

- There was no effective mechanism for ensuring a high level of knowledge. No competent outside organization periodically determined in-depth operator knowledge nor did the licensing and requalification process accomplish this. Unless the utility was enlightened and had the resources to ensure its own high standard, its operators might attain only mediocre knowledge and skill.

- There was no consideration given to training engineers at a higher level than the reactor operators. This stemmed, perhaps,

> from the underlying assumption that a nuclear reactor which produces power from a highly complex process and has the potential for affecting the health and welfare of the public can be operated solely by a few high school graduates or "equivalent."

The safe, efficient production of electricity by a nuclear reactor plant requires adequate numbers of properly trained and qualified operators in control. But if the operators are not properly trained and qualified then safe operation must depend on the plant design alone. Technology has not reached the point where the operator could be eliminated; therefore, training, qualification, and adequate staffing are essential to public safety and efficient electrical generation.

II. INTRODUCTION

There is evidence that action or inaction of operators, in the face of symptoms which indicated a loss of coolant, core uncovering, and high temperatures in the core and then core damage, failed to place the plant in a safe condition and did not isolate radioactivity from the environment. By the same token, managers who were apprised of the same symptoms did not understand their significance. Many actions taken may have exacerbated rather than mitigated the accident. These circumstances suggest that neither the operators nor their supervisors recognized what the symptoms indicated and did not understand the effects of many of their actions. If the events which took place in the Three Mile Island Unit 2 (TMI-2) control room were not the result of malfeasance, and there is no evidence to suggest that they were, then it can be reasonably postulated that the root cause for operator errors was inadequate training.

The purpose of this paper is to examine the selection, training, qualification, and licensing of operators in some detail. The term "operator" should be understood to include all those persons involved in reactor operation including nonlicensed operators, reactor operators, supervisors, engineers, and managers. The basic approach will be to list the requirements, describe how these requirements were implemented, and then evaluate the results in terms of the implementing programs. The nonexistence of needed requirements also will be discussed.

Preparation of this paper involved examination of many documents of both a prescriptive and descriptive nature, records of training conducted by Babcock & Wilcox and Met Ed as well as NRC Operator Licensing Branch records. Every available record pertinent to this area and known to the investigator was reviewed. Interviews of Babcock & Wilcox Company training services personnel were conducted on June 19-20, 1979. This was followed up by deposition of the same persons between July 3-6, 1979. Met Ed training department members were interviewed on June 26-27, with depositions being taken on July 27-31. NRC Operator Licensing Branch interviews and depositions took place on July 23 and July 28, respectively. All interviewees and deponents were very helpful. Requested documents were promptly produced by the organization from which they were sought. No particular difficulties were encountered in this investigation.

III. ANALYSIS OF SELECTION, TRAINING, QUALIFICATION, LICENSING, AND STAFFING

REQUIREMENTS FOR SELECTION, TRAINING, LICENSING, AND STAFFING OF REACTOR OPERATING PERSONNEL

The Atomic Energy Act of 1954 established the requirement that all persons who operate nongovernment-owned nuclear reactors must be licensed. Section 107 of the act established a requirement that uniform conditions for licensing individuals as operators be prescribed. In addition, the Atomic Energy Commission (AEC) was required to determine the qualifications of such persons and issue licenses to individuals in such form as the Commission may prescribe.

Title 10, Chapter 1, Part 50 of the Code of Federal Regulations (10 CFR 50), Licensing of Production and Utilization Facilities, paragraph 50.34(b)(6)(i), requires that the application for a license to operate a facility include information concerning the "applicant's organizational structure, allocation of responsibilities and authorities, and personnel qualifications requirements." Part 50 requires that the licensee not permit manipulation of the reactor controls by anyone who is not a licensed operator or senior operator.1/ Additionally, Part 50 requires

> Within three (3) months after issuance of an operating license, the licensee shall have in effect an operator requalification program which shall, as a minimum, meet the requirements of Appendix A of Part 55 of this Chapter. Notwithstanding the provisions of Section 50.59 the licensee shall not, except as specifically authorized by the Commission, make a change in an approved operator requalification program by which the scope, time allotted for the program or frequency in conducting different parts of the program is decreased.2/

Finally, Part 50 requires that apparatus and mechanisms other than controls that may affect the reactivity or power level shall be manipulated only with the knowledge and consent of a licensed operator. An operator or senior operator must be present at the controls at all times during facility operation and a senior operator must be present during operation or readily available on call. The senior operator is required to actually be present in the facility during certain specified plant conditions including recovery from an unplanned shutdown.3/

The basic Nuclear Regulatory Commission (NRC) regulation which implements the requirements of the Atomic Energy Act of 1954 is 10 CFR 55. This regulation establishes procedures and criteria for the issuance of licenses to operators, including senior operators. With regard to the selection and training of persons to become operators, however, there is no regulation; the NRC has no minimum eligibility requirements for either operators or senior operators and has not prescribed any training standards for the qualification of operators. Rather, the NRC in a part of a regulatory guide to the nuclear industry, endorses a standard established by the American Nuclear Society (ANS) pertaining to selection and training of nuclear power plant personnel.

Regulatory Guide 1.8, Personnel Selection and Training, describes a method acceptable to the NRC staff of implementing the portion of the NRC's regulations (10 CFR 50) with regard to personnel qualifications. Subcommittee ANS-3, Reactor Operations, of the American Nuclear Society Standards Committee developed a standard containing criteria for the selection and training of nuclear power plant personnel. Subsequently, it was approved by the American National Standards Institute (ANSI) Board of Standards Review and designated ANSI 18.1-1971, Selection and Training of Nuclear Power Plant Personnel. Regulatory Guide 1.8 states that the criteria for the selection and training of nuclear power plant personnel contained in ANSI 18.1-1971 are generally acceptable and, with one minor exception, 4/ provide an adequate basis for the selection and training of nuclear power plant personnel. As noted by Government Accounting Office (GAO) report, EMD-79-67, dated May 15, 1979, "the standard is intended as a guide and does not preclude anyone from not conforming to it."5/

Standard ANSI 18.1-1971 provides criteria for the selection and training of personnel for stationary nuclear power plants and addresses itself to the qualifications, responsibilities, and training of both operating and support personnel. The standard includes the following recommendations:

- that managers, supervisors, operators, technicians, and repairmen have prescribed levels of education, experience, health, and skill;

- that desired experience may be gained through actual nuclear power plant operation, including military, nonpower plant nuclear experience, simulator training, training programs, or on-the-job training;

- that a training program be established and maintained to provide adequate numbers of fully trained and qualified operating, maintenance, professional, and technical support personnel;

- that candidates for NRC cold examinations be qualified by a combination of experience and technical training; that candidates for NRC hot examinations complete certain technical training, on-the-job training, practice startups and shutdowns, and programs of self-study and counseling;

- that a program be implemented for training personnel not requiring licenses;

- that retraining and replacement training be provided;

- that minimum health requirements be established; and

- that minimum education and experience levels be established, as follows:

-- The operator shall have a:

high school diploma or equivalent and

two years of power plant experience or its equivalent provided that a minimum of one year is at a nuclear power plant.

-- The senior operator shall have a:

high school diploma or equivalent and

four years of responsible power plant experience. A maximum of 2 years of the remaining 3 years of power plant experience can be fulfilled by academic or related technical training on a one-for-one basis.

-- The plant manager shall have acquired the experience and training normally required by the NRC for a senior reactor operator license whether or not the examination is taken.

-- The operations manager shall hold a senior reactor operator license.

-- No other management personnel need hold licenses.

The standard remains silent with regard to reliability, stress fitness, psychological screening, and criminal records.

In January 1978, the American National Standards Institute approved Revision 1 to 18.1-1978. The revision provides more detailed guidance reflecting additional experience with staffing and training nuclear power plant personnel. Major changes concern the requirements for qualifications of operating personnel and expansion of the requirements of the training program for licensing and requalification and for general employee training. In February 1979, a proposed Revision 2 to Regulatory Guide 1.8, which would recognize ANSI/ANS 3.1-1978, was issued for comment. It was not expected at the time of issuance that implementation by the NRC staff would take place before October 1979. The current status of ANSI/ANS 3.1-1978 is unclear.6/

Specific requirements for TMI-2 are contained in the Safety Evaluation Report (SER), the Final Safety Analysis Report (FSAR), and the plant operating license.

The SER, Section 13.0, stated that a minimum shift would consist of at least five persons of which one would hold a senior operator license and two would be licensed operators. Overall training would be the responsibility of the TMI-2 plant superintendent. The program for training and qualification would conform to ANSI 18.1-1971. The SER described a program by which TMI-1 supervisory personnel would obtain licenses for TMI-2' as well as a program by which reactor operators for TMI-2 would be selected from senior qualified auxiliary operators assigned to TMI-1. Finally, the SER stated that requalification training and replacement training would conform to 10 CFR 50, 10 CFR 55, Appendix A, and follow the guidance in ANSI 18.1.

The TM1-2 FSAR, Section 13.2, contains requirements for the staffing qualification and training of operators and members of the staff organization. In particular, the FSAR requires a minimum shift composition during reactor operation of one senior operator, two operators, and two nonlicensed personnel. At least one licensed operator must be in the control room when fuel is in the reactor and at least two licensed operators must be present in the control room during reactor startup, shutdown, and during recovery from reactor trips. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI 18.1-1971 for comparable positions, except the supervisor of radiation protection and chemistry, who shall meet or exceed the qualifications of Regulatory Guide 1.8. A retraining and replacement training program for the unit staff shall be maintained under the direction of the director for generation training and shall meet or exceed the requirements and recommendations of ANSI 18.1-1971 and Appendix A of 10 CFR 55.

Section 13.2.2 of the FSAR describes in some detail the requirements for the Met Ed operator requalification program. This program, which includes preplanned lectures, on-the-job training, annual evaluation examinations, and accelerated requalification for those requiring remedial work, is implemented by Administrative Procedure 1006 and is evaluated below.

IMPLEMENTATION OF REQUIREMENTS

Nuclear Regulatory Commission

Implementation of the requirements contained in 10 CFR 55 and Regulatory Guide 1.8 is the responsibility of the Operator Licensing Branch (OLB) of the Office of Nuclear Reactor Regulation. OLB is assisted in certain aspects of fulfilling these requirements by the Office of Inspection and Enforcement (I&E).

The chief, Operator Licensing Branch, reports to the assistant director of the Division of Project Management for Quality Assurance and Operations. The Division of Project Management, in turn, is a section of the Office of Nuclear Reactor Regulation. Thus, the OLB could be descibed as a sixth echelon organization in the NRC with the following basic function:

> Examines and licenses candidates for reactor operator and senior reactor operator license; develops qualification requirements, testing techniques, and standards for evaluation of candidates and conducts safety evaluation for design and operation of reactor projects.7/

The Office of Inspection and Enforcement, through its regional offices, is charged with conducting periodic inspections of nuclear reactor operating staff training. Inspection objectives as described in Chapter 4100 of the Inspection and Enforcement Manual are to:

- confirm that the licensee has trained the operating staff;
- confirm that a continuing program of training is being conducted; and
- verify that replacements receive training or have the experience equivalent to that required for originally selected personnel.

This is accomplished by conducting a semi-annual spotcheck of the staff training program to include attending training sessions, lesson plan review, training record review, and interviews of selected individuals in various job classifications. The inspection should review the licensee's training program and gauge its adequacy vis-a-vis the guidance in ANSI 18.1-1971 and the requirements of the Final Safety Analysis Report. Additionally, the inspector is required to review the licensee's qualification training program to verify that it is being conducted in accordance with regulatory requirements. 8/ However, the Operator Licensing Branch is responsible for reviewing the adequacy of the technical contents of written examinations administered to licensed operators during separate audits.

The principal document utilized by OLB for administering the requirements for operator licensing (as distinguished from training) is the NRC Operator Licensing Guide (NUREG-0094). This guide describes the procedures and criteria for the issuance of operator and senior operator licenses and is intended to assist applicants and facility licenses to better understand the pertinent provisions of the NRC regulations as they relate to operator licensing. Although the preface to the licensing guide emphasizes that compliance with its provisions is not required, the guidance is comprehensive and in some areas is even prescriptive. Therefore, its contents will be discussed in some detail.

NUREG-0094 consists of sections on how to prepare the license application, scheduling of examinations, content of operator and senior operator written examinations, operating tests, waivers, operating tests before criticality, license expiration, renewal applications, and denial of applications. Appendices include sample medical history forms, typical sample written examination questions, procedures for examination with no reactor startup demonstration, and an oral examination checklist.

An application for license must include the education and pertinent experience of the applicant. Certification that he has completed training as required by the facility licensee and that he has competently and safely operated the controls must be included. Application certification should list details on the courses of instruction administered by the facility licensee including the number of course hours, hours of training, nature of the training, and startup and shutdown experience received. Applicants must have manipulated the controls of the reactor through at least two reactor startups and participated in several other transients or have completed an approved training program using a simulator to meet the manipulation requirements. 9/ The need for the applicant to be licensed must be justified.

Written and operating examinations are scheduled well in advance; applications should be submitted at least 2 months before the expected date of examination administration. The topical content of the operator written examination is set forth in Section 55.21 of 10 CFR 55. The 12 topics listed in 10 CFR 55 are rearranged by NUREG-0094 into seven categories for written examinations:

1. Principles of Reactor Operation

2. Features of Facility Design

3. General Operating Characteristics

4. Instruments and Controls

5. Safety and Emergency Systems

6. Standard and Emergency Operating Procedures

7. Radiation Control and Safety

The senior operator written examination includes the following additional topics:

8. Reactor Theory

9. Radioactive Material Handling, Disposal, and Hazards

10. Specific Operating Characteristics

11. Fuel Handling and Core Parameters

12. Administrative Procedures, Conditions, and Limitations

Evaluation of the content of operator and senior operator written examinations expected by both 10 CFR 55 and NUREG-0094 indicates that an elementary rather than comprehensive level of knowledge of nuclear reactor theory, design, and operation is sufficient. There is an emphasis on memorizing numbers, on how parameters change, and on how manipulations are performed. With regard to standard and emergency operating procedures, the operator is required to "demonstrate complete understanding of the symptoms, automatic actions and immediate action steps"10/ but there is no indication he need understand the principles of the operating and emergency procedures. There is a fundamental lack of emphasis on applying the theoretical to the practical. Required examination content for operators does not include the demonstration of a comprehensive knowledge of the following:

° plant materials, stress, brittle fracture

- reactor coolant system (RCS) chemistry
- secondary system chemistry
- reactor core thermodynamics
- overall plant thermodynamics
- emergency thermodynamics including decay heat, core cooling, and natural circulation
- saturation phenomena
- radiation theory
- hazards to the public including the amount of radioactivity which could be released and the potential consequences of such releases
- loose surface contamination
- airborne radioactivity
- personnel surveys and decontamination
- interrelationships between the secondary and primary systems
- emergency situations not covered by emergency procedures

As a part of the licensing process, 10 CFR 55 requires that the candidate be administered an operating test. Among other things, the applicant is required to conduct a reactor startup from shutdown to power. NUREG-0094, on the other hand, permits an alternative program, not recognized by the regulations. If the applicant satisfactorily completes an NRC-approved training program consisting of a 500-hour lecture series on subjects listed in ANSI 18.1-1971 and a program of at least one week's duration at a nuclear power plant simulator, no actual reactor startup needs to be performed during the license examination. 11/ This program was initiated because of the burden imposed on the utility of requiring an actual reactor startup.12/ The senior operator and operator operating tests are similar in scope. The senior operator need demonstrate a greater depth of knowledge, but he is not intended to be nor is he tested as a supervisory reactor operator.13/

If the applicant has had extensive actual operating experience at a comparable facility within 2 years prior to the date of application and has discharged his responsibilities competently and safely, the NRC may waive any or all the requirements for a written examination and operating test provided that the facility licensee certifies that the applicant is qualified to operate the facility.

NUREG-0094 gives procedures for implementing the provisions of 10 CFR 55 concerning the administration of operating tests prior to initial criticality. These examinations are termed "cold" examinations as opposed to examinations given at an operating reactor, which are called "hot" examinations. A cold examination is given if there is an immediate need for the services of the operator or senior operator license applicant and the applicant has had extensive actual operating experience at a comparable reactor. NRC considers for this purpose that any light water power reactor is comparable to any other light water power reactor. It also considers that completion of an NRC-approved simulator program is extensive actual operator experience.

NUREG-0094 also provides details of application denial and reapplication. If an application is denied because of failure to pass the written examination, operating test, or both, the applicant may reapply 2 months after the date of denial. A third application may be filed 6 months after the second denial. Subsequent applications may be submitted at 2-year intervals.14/

Appendix E to the Operator Licensing Guide includes sample questions typical of those appearing in operator and senior operator written examinations.

Appendix G to the Operator Licensing Guide is a sample Examination Report (NRC Form 157) which serves principally as a checklist for the operating test.

Metropolitan Edison Program

General

Training conducted at Three Mile Island is primarily intended to result in the successful completion of NRC licensing requirements for control room operators (CROs) and senior reactor operators (SROs) and then to ensure that the requalification program requirements of Appendix A to 10 CFR 55 are met. Secondary objectives include providing training for persons not requiring licenses and general employee training as specified in ANSI 18.1-1971.

Training to ensure that candidates for NRC operator licenses are properly qualified includes auxiliary operator training and either "cold" or "hot" operating licensing programs. The terms "cold" or "hot" refer to whether the license is being obtained prior to initial core fuel loading or subsequent to initial criticality. The training of replacement operators is also accomplished by means of the "hot" licensing training program. Operator proficiency and certification are maintained through the licensed operator requalification program and periodic examinations. The manner in which these programs are intended to be carried out follows.

The normal progression of qualification commences with auxiliary operator C and continues through auxiliary operator (AO) B and A. An AO'C' operates equipment only under the supervision of a more senior operator. AO'B's are permitted to operate secondary systems and equipment without supervision whereas an AO'A' is allowed to operate both primary and secondary equipment outside the control room. Auxiliary operator A can apply for ("bid on") the position of CRO trainee when such a position becomes vacant. Licensed CROs can compete for available SRO positions.

Auxiliary Operators (AOs)

The program for the training and qualification of auxiliary operators is not defined formally. Consonant with the recommendations of ASI 18.1-1971 regarding qualification of nonlicensed operators, Metropolitan Edison requires all auxiliary operators to possess a high school diploma or equivalent, with additional educational attainments for progressively higher levels.

Auxiliary operator C must have completed a course in algebra. They are required to attend a 9-week classroom training program which covers the following areas:

- secondary plant systems
- fundamentals of mechanics and heat transfer
- the basic steam cycle
- health physics
- switching and tagging procedures
- basics of safety features actuation system
- basics of reactor protection system
- first aid
- fire fighting

Upon completion of this initial training, AO'C's must pass a written examination. They are then assigned to a shift and work under the supervision of an AO'A' for the remainder of a year. In order to become an AO'B', the AO'C' must pass another written and oral examination. If he fails the examination he is removed from the training program.

At the AO'B' level trainees must have completed courses in high school physics and trigonometry. They are required to complete a 6-week classroom program which is culminated by a written examination. Topics included are:

- make-up system
- decay heat removal
- core flood system
- reactor building cooling and spray
- study of the safety features actuation system and the reactor protection system

- nuclear instrument system
- control rod drive mechanisms
- integrated control systems
- advanced health physics

For the remainder of the one-year period as an AO'B', the operator is engaged in on-the-job training under the supervision of a shift foreman. At the end of the year, the trainee must pass both written and oral examinations in order to remain in the program and to be promoted to AO'A'. Once promoted to AO'A' he is then eligible to become a control room operator trainee when such a position becomes available.

"Cold" License Training Program

As provided for in Section 5.2.1 of ANSI 18.1-1971 and Section 13.2 of the TMI-2 Final Safety Analysis Report (FSAR), Met Ed developed a program for "cold" licensing of the initial TMI-2 staff. This program was reviewed by the NRC Operator Licensing Branch for compliance with established standards and was formally approved. Section 13.0 of the Safety Evaluation Report concluded that the training program and schedules for TMI-2 staff members were acceptable for the preoperational test program, for operator licensing examinations, and for fuel loading. Although this was a one-time program for the purpose of ensuring properly trained and licensed operators would be at the controls for initial fuel loading and reactor startup, it is important that its significance be understood in the overall process of operator licensing. Therefore the "cold" licensing program for TMI-2 will be described briefly.

Met Ed's program for initial TMI-2 staff "cold" licensing was tailored to the needs of the reactor operators of TMI-2 who were to be selected from senior qualified auxiliary operators assigned to TMI-1. Their training was to include approximately 200 hours of classroom training, training in the differences between TMI-1 and 2, and a one-week course at Pennsylvania State University on core physics and reactor startups. One month was to be spent in observation training on TMI-1 and eight weeks training on the Babcock & Wilcox simulator. Finally, the program was to include both written and oral examinations similar in nature to those administered by the NRC for cold licensing purposes.

"Hot" License Training Program

"Hot" license training is correctly referred to as replacement operator training. At TMI the program for preparing replacement operator candidates for "hot" licenses is known as the Category IV Training Program. The Category IV Program has not been sanctioned officially by means of an administrative procedure. Rather, the promulgating document was Training Department Administrative Memorandum Number Five, dated Oct. 8, 1976. This memorandum was submitted to the station superintendent, the unit superintendents, and all department heads.15/

When an AO'A' bids on a control room operator position and is advanced to Category IV control room operator (CR0), he will be placed in a 9-month training program consisting of specific study assignments, practical work, written tests, oral examinations, and classroom sessions. Throughout most of this period, the individual will be on shift.

The replacement operator program is intended to provide in-depth coverage of all areas specified in ANSI 18.1-1971 and the TMI-2 FSAR, namely:

- reactor theory
- features of facility design
- general operating characteristics
- instrumentation and control
- safety and emergency systems
- standard and emergency operating procedures
- radiation control and safety

Administratively, the Category IV Training Program is to be handled in the following fashion. Upon being advanced to CRO, the individual will be assigned to a shift. Two hours, as a minimum, of each day on shift will be specifically devoted to training. The individual will be provided with a desk or other suitable place to study in the control room area. While on shift, the individual receives a series of preprogrammed written assignments. He is administered written and oral examinations every 3 and 6 weeks, respectively. Errors and weak areas on both written and oral exams will be reviewed with the individual; failure of a written exam or oral exam will be discussed with the individual and a retest will be administered on the material. Additionally, the CR0 will be required to complete a practical evolutions sheet. This sheet will be completed either during his daily training period or during other times while on shift as situations dictate. Most of the items involve performing evolutions, simulating performing evolutions, and understanding and being able to explain while simulating or performing. The individual's shift supervisor, shift foreman, an SRO-licensed individual, or the licensed training coordinator may sign the practical evolution sheet. Assignments on which written and oral tests will be given will come largely from items on the practical evolution sheet. Checkouts for items on the practical evolution sheet that must be simulated will be conducted in front of the control room consoles and panels with the trainee being required to point to specific items and controls. The checkout must be satisfactory prior to a signature for the evolution. The evolutions are assigned a point value to track the progress of an individual through the program.

To aid the individual in the training assignment completion, he my come off shift to attend lectures on specific topics, listed below, as determined by the supervisor of the training department and the supervisor of operations.

Area	Duration
reactor theory	1 day to 1 week
integrated control system	1 day to 1 week
simulator	1 week or 2 weeks 16/
health physics review	1 day to 1 week
refueling review	1 day to 1 week

These off-shift lectures are intended to aid the individual in obtaining signatures on the practical evolution sheet. They are also the manner in which the trainee meets the requirement for 500 hours of lectures specified in Appendix F to NUREG-0094.

The first 90 days of the CR0 training program are designated as a probationary period during which the individual will be evaluated. At the end of this 90-day period, the shift supervisor, supervisor of operations, and the supervisor of training will recommend whether or not the individual should continue in the program.

Prior to the completion of the 9-month period, the CR0 will be given a comprehensive written examination approved by the supervisor of operations and the supervisor of training, and the CR0 will receive a comprehensive oral examination by an SRO-licensed individual designated by the supervisor of operations.

If the CR0 has not successfully completed the program within 9 months or fails either the written or the oral examination, he will return to the position held prior to being advanced to CRO. If the individual successfully completes the training program within 9 months and fails either the written or oral examination, a re-examination will be considered based upon an evaluation by the supervisor of operations and the supervisor of training. If the trainee successfully completes the training program within 9 months and passes the final comprehensive written and oral examinations, he may be recommended for examination by the NRC and subsequent reactor operator licensing.

Shift Foreman Training and Certification

Certification as a senior reactor operator (SRO) is accomplished also by means of satisfactory completion of NRC examinations. Training leading to SRO qualification is achieved through programs which, again, have not been officially established by means of an administrative directive.

The <u>de facto</u> training, qualification, and certification program is tailored to suit persons selected from one of three backgrounds:

- individuals who had achieved and maintained SRO licenses on TMI-1;

- individuals who had achieved SRO licenses on other facilities; or

- individuals selected from the initial group of TMI-2 control room operator trainees.

Training is expected to include formal classroom instruction, Babcock & Wilcox simulator training, in-plant experience, and company-administered examinations in preparation for the NRC licensing examinations. The details and relative emphasis will be varied not only within the three categories of backgrounds but also in individual cases.

Shift Supervisor Training and Certification

Shift supervisors are required to be certified at the senior reactor operator level for both TMI-1 and TMI-2 through satisfactory completion of NRC-approved examinations. For initial dual-unit staffing, a Cross License Training Program was administered to obtain TMI-2 SRO licenses. Classroom, on-the-job, and simulator training were conducted. At the conclusion of the program, a mock NRC examination was administered with emphasis on TMI-2 systems and the differences between the TMI-1 and TMI-2 nuclear steam supply systems, secondary systems, and balance of plant systems. The program documentation, exam, and exam results were forwarded to the NRC for approval. This culminated in persons already licensed in TMI-1 having their licenses amended to include TMI-2.

Subsequent to "cold" licensing, a "hot" license training program was developed by the training department to cross qualify SRO license holders from either unit. Again, this program has not been officially defined. It is intended to be a self-study course with periodic written and oral examinations to monitor the individual's progress. As in the case of the initial cross licensees, this program culminates in a written examination administered by the training department. The results are reviewed by the NRC and amendments to licenses issued to include the other unit.

Requalification Program

Pursuant to the requirement of Appendix A to 10 CFR 55 that each licensed operator demonstrate his continued competence, Met Ed established a Requalification Program by means of Administrative Procedure 1006. This procedure was submitted to the NRC as a portion of the Final Safety Analysis Review.

The Met Ed Requalification Program consists of four interrelated segments which run concurrently. These are:

- Operational Review (OR) Lecture Series;

- Fundamentals and System Review (FSR) Program;

- on-the-job training (OJT); and
- annual evaluation examinations.

The OR series is a classroom lecture presentation which provides licensed personnel with the details of operational information. As a part of the OR series, FSR topics selected from areas of the annual written examinations in which operators had difficulty are given. OR lectures are scheduled for a minimum of 60 hours per year. On-the-job training is intended to ensure all licensed persons operate the reactor controls and participate in major plant evolutions. The annual evaluation examinations simulate the written and oral NRC examinations. Performance on these annual evaluation examinations determines the extent of the FSR program during the following 12-month period. Each licensed operator must complete all OR and FSR requirements on an annual cycle. On-the-job training is conducted throughout the 2-year term of the individual's license. A statement of requalification program participation will be submitted with each license renewal application.

Operational Review Lecture Series attendance is required of all licensed operators and senior operators on shift assignments. The following topics will be covered as a minimum each year:

- reportable occurrences
- unit modifications
- operating history and problems
- procedure changes
- abnormal and emergency procedure review
- technical specifications
- major operational evolutions
- applicable portions of Title 10, Chapter I, CFR
- FSR program material

Additional topics which may be covered include:

- operational Q/A
- standing orders
- operating experiences, reactor safety, and pertinent NRC publications

Absences will be made up by reviewing lecture materials and/or discussions with on-shift supervisors or technical staff. Periodic quizzes covering content of OR lectures will be administered.

The Fundamentals and Systems Review Program participation for licensed persons is based on their annual written examination scores. The FSR program may consist of lectures, self-study, tutorial sessions, and quizzes.

On-the-job training ensures that each licensed operator (CR0 or SRO) participates in at least 10 reactivity manipulations; participates in applicable surveillance testing, system checkout, and equipment operations; and reviews procedure changes, equipment modifications, and technical specification changes. Diversity of operations is required. Licensed personnel, whose job assignments are not directly related to unit operations, will actively participate in control room operation 48 hours per year. This may be performed in a simulator.

The annual written evaluation examination will be administered to all licensed operators and will simulate the NRC written examinations. The annual oral examination will be administered to all licensed persons using a checklist; prescribed areas will be covered.

Persons scoring less than 80 percent on any section of the annual written examination will be required to attend the FSR program related to failed sections.

Babcock & Wilcox Program

The Babcock & Wilcox (B&W) Training Services Section, a part of the Customer Services Department, provides training services to utility customers to assist in the training, qualification, and licensing of reactor operators and senior reactor operators and to train management personnel and engineers in the fundamentals of B&W-designed nuclear steam supply systems (NSSS). Additionally, there are some specialized, highly technical courses available.

The Training Services Section does not have a formalized training program. Rather, the company offers a variety of individual training courses which the customer may contract for and which are described only briefly in a Nuclear Training Services Catalog. These courses are as follows:

Course Number	Course Title
T101	Management Seminar
T102	Basic Nuclear Orientation for Managers
T103	Nuclear Power Plant Operations for Management
T201	Engineering Staff Orientation
T202	Plant Operations for Engineers

T203	Physics Test Orientation for Engineers
T301	New Plant Operator Program
T302	NSSS Orientation for Experienced Engineers and Operators
T303	Replacement Operator Training
T304	Simulator Requalification Training
T401	Nuclear Plant Maintenance
T402	Instrumentation, Control and Computer System Training
T403	Chemistry Technician Training
T501	NSSS Videotape Seminar

The most valuable service which B&W can provide is to offer "hands-on" training to operators and operator candidates in manipulating simulated reactor controls during normal and emergency conditions. As far as the NRC is concerned, the manipulation of controls and the simulation of emergency or abnormal conditions may be accomplished by using the control panel of the facility or by using a simulator. Since the use of a simulator is encouraged 17/ and because it is not practical to perform power level changes and casualty drills for training on an operating commercial power reactor, the B&W simulator is used extensively for operator training.

The most commonly used courses at B&W and the only courses contracted for by Met Ed for its operators are T301, T303, and T304. Only these will be described here.18/

New Plant Operator Training / T301

This course provides utility personnel with the training necessary to become reactor plant operators. The program has been certified by the NRC to meet all prerequisites for the "cold" license operators to support an initial plant startup. The complete program consists of five different courses designed as a package to fulfill current NRC "cold" license requirements. Included are courses in basic nuclear theory (12 weeks), observation experience (8 weeks), simulator operations (8 weeks), nuclear steam systems (4 weeks), and on-the-job training at the customer's site (10 months). The most commonly used course is the simulator operations course consisting of practical instruction on the full-scale B&W pressurized water reactor (PWR) simulator, related classroom instruction, and individual study time. The program consists of 2 weeks of instruction in the classroom, 5 weeks of simulator operation, and one week of NRC-type written and operational examinations.

Trainees are instructed in the control room in groups of three, with every student operating the simulator at each of the three operating positions. Emphasis is on operational orientation with the trainee concentrating on learning the basic plant operations, casualty procedures, performing assigned evaluations, and handling improved casualties. Lectures reviewing plant systems from an operational and functional viewpoint are used to supplement simulator operation. Plant operating procedures are presented as a planned sequence coinciding with actual operations on the simulator.19/

Replacement Operator Training / T303

The purpose of this program is to provide training for "hot" license candidates in plant operations by operating the B&W simulator. The two-week course is evenly divided between the classroom (40 hours) and the simulator (40 hours). The method of instruction is classroom discussion followed by practical demonstrations on the simulator. Simulator exercises provide the student with experience in controlling normal and emergency plant evolutions with emphasis on operation of the Integrated Control System (ICS). This course also includes an NRC-type startup examination on the simulator.20/

Certification is made to the utility management that students have satisfactorily completed the startup examination.

The typical course schedule is as follows:21/

Day 1
- Introduction
- Control panels
- Startup procedures
- Reactor criticality

Day 2
- Reactivity balance calculations
- Plant shutdown (hot shutdown to 25 percent power)
- Plant startup (hot shutdown to 25 percent power with turbine generator in operation)

Day 3
- Technical specifications related to startup
- Power operations and major malfunctions

Day 4
- Review of startup procedure
- Reactor startup practice

Day 6
- Integrated control systems (ICS) operation
- Manual/automatic ICS power operation including turbine and reactor trips

Day 7
- Engineered safety features actuation system review
- Reactor coolant system leaks
- Reactor coolant pump and/or feedwater pump trips

Day 8
- Steam leaks, turbine by-pass valve and feedwater subsystem operation
- Turbine by-pass valve and feedwater subsystem failure

Day 9
- Control rod drive operation
- Control rod drive malfunction and instrument failures

Day 10

Review and power operation with unannounced casulaties

Simulator Requalification Training / T304

The Simulator Requalification Training Program provides assistance to the utility in meeting operator on-the-job requirements for an operator requalification program as specified by 10 CFR 55, Appendix A. The program includes a review of recent abnormal occurrences applicable to B&W plant operations as well as a review of the utility's abnormal and emergency procedures. Instruction consists of 20 hours of practical exercises on the B&W simulator and 20 hours of classroom instruction.

A typical course schedule follows:22/

Day 1

Introduction
Review of procedures; reactor startup and control rod drive operation
Control room orientation
Practical exercise: plant startup (from all rods in to 20 percent power)

Day 2

Review of procedures: general power operations
Integrated control system reviews
Control rod drive mal-functions
Review of operating curves and limits
Practical exercise: plant startup (1 percent shutdown to 100 percent power), integrated control system operation in auto and manual, controlrod drive malfunctions

Day 3

Review of technical specifications
Evaluation of leak rates
Practical exercise: reactor coolant, system rupture, steam generator tube leaks

Day 4

Review of procedures: reactor trip, turbine trip, steam rupture
Review of selected transients and plant response
Practical exercise: power operations, manual intergrated control system operations, instrumentation failures

Day 5

Review plant response to selected nonnuclear instrumentation failures
Review of safety analysis and reactor protective systems set points
Practical exercise: power operations, manual integrated control systems operations, instrumentation failures

Facilities and Equipment

The B&W Nuclear Training Center occupies approximately 4,000 square feet of space and includes two classrooms, two study rooms, a technical reference area, and a pressurized water reactor (PWR) simulator.23/

Each of the two classrooms is designed to accommodate 24 students. The classrooms are equipped with self-contained audiovisual aid systems. A student response system is utilized for rapid and continuous monitoring of comprehension of classroom presentations.

The PWR simulator is modeled after the Rancho Seco nuclear generating station control room. It represents the B&W 177 fuel assembly, lowered loop nuclear steam supply. The consoles are essentially the same as those in control rooms of all B&W-designed nuclear power plants.

The main consoles of the simulator are electronically coupled to a digital computer which is programmed to simulate the dynamic performance of the actual plant. Movement of control devices by the student initiates responses from the computer which, in turn, sends back signals to the consoles for moving dials, energizing lights, and actuating alarms.

The following systems are included in the simulation:

- reactor core
- control rod drives
- reactor coolant system
- steam generators
- steam, condensate, feedwater
- turbine generator
- let-down and make-up
- deboration
- chemical addition
- decay heat
- all emergency injection
- intermediate cooling
- raw water cooling
- radiation monitoring
- instrumentation
- protection and safeguards
- reactor building
- electrical
- automatic control

In addition to the dynamics of these systems, the simulation includes all logic operations associated with mode change of valves, motors, and signals.

Additional realism can be created by duplicating and/or simulating the annunciator panels, communications, control room light fluctuations, and background noise.

The instructor's control room and console is located adjacent to the simulator control room. From this console, the instructor can perform the following functions:

- ready the plant for training operations at any one of 16 different conditions, including subcritical, critical, various power levels, various stages of core burnup, various poison levels, and various temperature and pressure levels;

- act as load dispatcher for the utility's power network which involves setting the electric load demand to the simulated plant;

- act as "outside plant assistant" in performing certain operations which are not designed for remote control at the main consoles; and

- introduce any one of approximately 80 simulated plant malfunctions. Most of these malfunctions are not simply "on-off" type, but rather can be the type that can only be controlled by proper operator action.

Actual duplicates of the integrated control system hardware are installed with the simulator and interfaced with other components.

A "freeze" feature is included whereby all computed variables are held constant and then released on separate commands from the instructor. All console meters will hold while the instructor emphasizes or reviews some critical point in a transient.

EVALUATION OF OPERATOR, SELECTION, TRAINING, LICENSING, AND STAFFING

Nuclear Regulatory Commission

Requirements and Implementing Procedures

As was discussed in the first part of this paper, the requirements for operator licensing are contained in 10 CFR 50 and 10 CFR 55. There are no requirements for operator selection and training; guidelines are given in Regulatory Guide and ANSI Standard 18.1-1971. Further guidance for licensing is contained in the NRC Licensing Guide.

The aggregate of these regulations, guides, and standards form a body of procedures which by mutual consent and common understanding institutionalize the training and licensing of nuclear reactor plant operators. The requirements which do exist are very limited in scope and in many respects are vague with regard to intent. Part 50 of 10 CFR, which is concerned with facility licensing, makes few, general statements regarding the operators who will manipulate the reactor controls and their qualifications. Part 55 of 10 CFR, Operator Licenses, provides only a framework for the licensing process although it does include a more detailed description of requalification requirements in an appendix which was added in 1973.

Operator licensing regulations as a whole provide sufficient details as to the administration of operator licensing and qualification, but they are deficient in defining the scope and depth of theoretical knowledge and practical factors necessary to ensure competent reactor operators. The Licensing Guide, which need not be complied with, expands on the provisions of the regulations; this document, as well, does not provide adequate guidance. Significant deficiencies in regulatory requirements include the following:

- There are no requirements concerning the qualification of engineers and managers. These persons, who would likely be called upon in case of an accident to make significant decisions which could directly affect the outcome, are not required to be licensed. Those who do hold SRO licenses are not required to demonstrate a level of knowledge any more comprehensive than that of the reactor operators. Thus, a superior ability to rationalize an unusual set of circumstances or to deal with a situation not predicted by procedure writers would be based solely on past educational achievements rather than a comprehensive knowledge of the reactor plant. The "last line of defense," the engineer, is perhaps less qualified than the persons who turn to him for direction.

- The auxiliary operators, who operate apparatus and mechanisms that affect the reactivity or power level of the reactor, are not subject to any regulatory requirements concerning their qualifications or level of knowledge. The persons responsible for the operation of the condensate and feedwater systems of TMI-2, for example, were auxiliary operators.

- The regulations do not address any aspects of the licensing process other than the written examination and an operating test. Licensing examinations and tests, because of time constraints, can only be spot checks. There are no requirements in the regulations that trainees demonstrate a satisfactory level of theoretical knowledge in individual areas, as determined by qualified utility examiners, or that certain practical factors be performed. There are no prescribed areas in which a high level of theoretical knowledge must be attained. Part 55.21 includes areas in which questions will be asked on written licensing examinations, but the emphasis of these questions is on details of construction, operation, procedures, functions, and characteristics. Review of the content of operator written examinations and sample operator examination questions given in the Operator Licensing Guide, as well as questions which have been given in actual operator examinations, confirm that the regulatory philosophy is to concentrate on the "whats," "how manys," and "lists" and not on the "whys." For instance, 28 of the sample questions in the Licensing Guide begin with the word "what" and only two begin with "why." As far as content of the written examinations is concerned, there are no questions asked of reactor operators about the principles of decay heat production and removal, the importance of keeping the core covered, core cooling, thermodynamic principles, the principles of saturation, subcooling and superheat, radiation theory, or reactor kinetics. In summary, the regulations promote a shallow level of operator knowledge.

Operator Eligibility and Selection

The Nuclear Regulatory Commission has no requirements concerning minimum education, experience, reliability, criminal record, or stress fitness. It merely endorses a guideline standard established by the American National Standards Institute in 1971 and does not require adherence to it. A follow-on standard, dated 1978, has not been endorsed although the NRC Operator Licensing Branch, B&W Training Department, and Three Mile Island Training Department are "kind of going by it."24/

As far as the guidelines are concerned, operators and senior reactor operators of nuclear power plants, which have the potential for great impact on the public health and safety, need only be high school graduates or equivalent. The term "equivalent" is nowhere defined. In fact, there are no firm educational guidelines. The chief of the NRC Operator Licensing Branch has stated that no one has ever been refused a license because of a lack of education. It is considered that if a candidate can make it through a utility training program he must not have any substantial educational deficiencies.25/

The guidelines recommend that an operator or senior operator license candidate have a minimum of one year of nuclear power plant experience. The term "power plant experience" is nowhere defined. Power plant experience can be gained prior to the time that a plant is operated at power, such as during the startup program. 26/ The OLB also gives credit for nuclear experience that a candidate may have gained in the Navy regardless of length of service, type of duty, rating (electrical, mechanical, reactor), or performance. 27/ In cold licensing, experience credit is given for all off-site training.28/

A license applicant must submit a certificate of medical examination attesting to his physical condition. 29/ It also includes a brief medical history prepared by the applicant. The applicant must indicate whether or not he/she has ever been treated for a mental or nervous condition and whether he/she has ever been rejected for or discharged from employment because of mental or nervous disorder. Medical examination for the NRC license application does not provide for any psychiatric screening nor is there any consideration given to the candidate's ability to respond in stressful situations.

Similarly, there is no investigation of an applicant's criminal history other than requiring him to indicate in his medical history statements whether he has ever been convicted of a law violation resulting in a fine of $25 or more. The Operator Licensing Branch does not verify any of the responses given in medical history certificates.30/

Training

Again, the Nuclear Regulatory Commission has no training requirements for either utilities or nuclear steam supply systems (NSSS) vendors who provide simulator services; provisions in ANSI 18.1-1971 concerning the training of licensed and nonlicensed operators as well as general employee training are only recommendations. There are no minimum training requirements.

NRC is involved in operator training programs to a limited degree. Specific actions taken are as follows:

- The initial training programs at Three Mile Island were developed by Met Ed and submitted to the NRC for a paper review and approval. Some advice was given by the NRC in setting up the cold licensing training program. 31/ The hot license (Category IV) training program and replacement operator training programs were given paper reviews by the OLB as they were developed. There have been no further reviews and the OLB has not conducted any audits of the Met Ed training program, other than requalification.32/

- The Met Ed requalification program is given spotcheck audits periodically by the Office of Inspection and Enforcement. This inspection, which is ostensibly for the purpose of verifying that the requalification training program is conducted in accordance with regulatory requirements,33/ is essentially a review of records. Inspectors neither monitor the conduct of lectures nor offer comments on the substance of material covered.34/

- The OLB was "heavily involved in the initial setup of the Babcock & Wilcox cold licensing program" in the early 1970s. This involved examining the first graduates and becoming satisfied that the program "would do the job." 35/ A paper review of B&W course T301 (New Plant Operator Training) was performed without comment in 1976.36/

- About every 6 months, an OLB representative has observed the conduct of a startup certification test at B&W. He has not provided a written report of his observations.

Other than these actions, the NRC is not significantly involved in the training of operators. Specifically, neither OLB nor the Office of Inspection and Enforcement (I&E) has any criteria for or involvement in the following areas:

- staffing or qualifications of instructors in the Met Ed and B&W training departments except that senior simulator instructors should have held SRO licenses at some time in the past (this is an unwritten requirement);

- qualifications of the head of either training department;

- in-depth audits of training conducted at either Three Mile Island or Babcock & Wilcox;

- content or conduct of simulator training programs for either replacement operators or requalification (in fact, simulator training is not even required by the NRC); 38/ and

- content or conduct of classroom training at either TMI or B&W, with exception of verifying that scheduled requalification lectures were held at TMI.

The OLB does not have any objection to utilities "teaching the test." In fact "mock" NRC examinations are encouraged and most utilities give them. Giving such "mock" exams reportedly lightens the workload of the OLB.39/

With regard to training its own people in reactor operations, the Operator Licensing Branch occasionally has sent members to B&W for special simulator courses. The last time that this was done was in 1973 or 1974. 40/ Although a course was being set up for I&E inspectors at B&W, as of the time of the TMI-2 accident none had attended such a course.

According to the chief of the Operator Licensing Branch, the NRC has not reviewed either training programs or instructor qualifications but rather goes by the end product, that is, the licensing examination results.41/

Operator Examinations and Tests

Reactor operator and senior reactor operator candidates must pass written examinations administered by the NRC in order to receive their licenses. The written examinations for reactor operators cover the seven categories required by 10 CFR 50 and require approximately 6 to 8 hours to complete. Senior reactor operator candidates must be examined on five additional categories which include reactor theory, radioactive material handling, specific operating characteristics, fuel handling, and administrative procedures. The SRO examination requires an additional 4 to 6 hours. Reactor operator and senior reactor operator examinations differ principally in that the senior categories are more difficult and delve into plant operation more deeply. The difference between these examinations appears to be ill-defined. Additionally, the examinations reflect the defects of the regulatory requirements which have been discussed above.

In addition to the written examination, candidates must pass an operating test which also requires about 4 to 6 hours to complete. The operating test consists of three parts: a theoretical oral examination, a talk through of normal and abnormal operations in the control room, and, finally, a walk through of the plant. Oral examinations for SRO candidates are optional. No such examination has been given at Three Mile Island.

Examiners are responsible not only for administering the examination and operating test but also for preparation and evaluation of the results. As of the time of the accident at Three Mile Island, the Operator Licensing Branch consisted of only eight examiners, seven of whom were degreed engineers and three of whom had been licensed at one time. 42/ These examiners are organized into three groups 43/ to facilitate the Branch's work.

Eight persons cannot effectively handle the licensing examination task for the entire United States. Therefore, the OLB has resorted to hiring consultants to work on a part-time basis to prepare, give, and evaluate operator license examinations and operating tests. At the present, the OLB has in its employ 22 part-time examiners: about half work full-time for national laboratories 44/ or the Hanford Reservation Site and about half are college professors. Few of the consultant examiners have had any experience with commercial reactors.45/

Examinations are prepared by the persons who will administer them. Questions are taken from a "bank" of about 400 questions from previous examinations or from the plant's Final Safety Analysis Report, radiation protection manual, emergency plan, licensee event reports (LERs), and so on. Examinations prepared by OLB members are not reviewed for content but rather they are passed around to ensure they are "givable." Examinations prepared by consultants are reviewed by an OLB group leader at least 2 weeks before the examination is to be given. There is no record kept of this review.46/

Examinations are administered at the plant site to a group of candidates. The examinations are graded in Washington or the consultant's home office. Thus, results are not available at the time the oral examination is given to permit probing weak areas in depth.

The passing grade for the NRC written license examination is 70 percent overall. A person can fail one or more categories of the examination and still pass overall. Additionally, a person who fails one or more categories but scores greater than 70 percent overall need not receive additional training in the unsatisfactory areas.47/

Review of statistics for the years 1975 to 1978 indicates that of 852 reactor operator examinations given, 88 percent of the applicants passed. Of 377 previously licensed applicants taking the SRO examination, 90.2 percent passed. Of 560 first-time SRO applicants, that is, those not previously licensed as reactor operators, 88.8 percent passed.48/

Three Mile Island candidates have fared better than average. Since 1974 at the reactor operator level there have been no failures of about 30 candidates for "hot" licenses. The failure rate of 55 persons seeking senior reactor operator licenses was 12 percent. Significantly, the failure rate of "upgrade SROs" persons already holding reactor operator licenses was zero. All those who were unsuccessful were "instant SROs."49/

In testimony before the Subcommittee on Energy and the Environment of the House Interior and Insular Affairs Committee on May 24, 1979, the Met Ed vice president for generation stated:

At the recent American Nuclear Society Symposium on Training at Gatlinburg, Tennessee, April 29 - May 2, 1979, J. Holman of the NRC issued statistics on NRC license failure rates. In the period of 1971-1974, the failure rate in the industry was 15.2 percent. For this same period, TMI failure rate was 12.1 percent. For the period of 1975-1978, the industry failure rate was 11.5 percent. For this same period, the TMI failure rate was 5.3 percent.

This would indicate that the effectiveness of the Met Ed training program was better than the industry average.

License Issuance

The chief of the NRC Operator Licensing Branch is the grantor of all reactor operator and senior reactor operator licenses. His decisions are not subject to any review except for denials which must be approved by his superior, the assistant director of the Division of Project Management for Quality Assurance and Operations. The chief of OLB uses a medical consultant to review each license application but it is he who grants all waivers and conditional licenses. Typical conditions include requiring the operator to wear eyeglasses, to receive more frequent medical examinations than normal, or to operate the reactor only with another operator present. 50/ Licenses are issued for a 2-year period but, unless an operator is unable to complete the requirements of the utility-administered requalification program, it is unlikely that he will ever again interact directly with the NRC. Under very unusual circumstances, the operators license could be revoked. Of the approximately 2,500 operators in the country, the NRC has suspended one operator license and required reexaminations of only six other operators.51/

Requalification

Licenses for both reactor operators and senior reactor operators are of 2 years duration. A license will be renewed by the NRC without examination provided that the operator's physical condition is good and:

- he/she has actively and extensively engaged as an operator, has discharged his duties competently and safely and is capable of continuing to do so; and
- he/she has completed a requalification program.

The requalification program is conducted by the utility and should be audited by the Office of Inspection and Enforcement (I&E) representatives for its quality. The NRC requires through 10 CFR 55 that a written evaluation examination be given. In addition, there is an informal requirement that an oral examination of each licensee also be conducted. The NRC has declared informally that if anyone gets less than 80 percent in a category of the written examination that person should go to a lecture on the subject. If a person gets less than 70 percent in the annual written exam or is unsatisfactory in his oral examination, he must go into accelerated training. If he is unsatisfactory in both written and oral exams, he may not perform duties associated with reactor control. If an operator fails the written exam, he may or may not continue as an operator depending on the utility's program.52/

The requalification program must also require each operator to manipulate the controls at least 10 times in a 2-year period. Such manipulation may be performed on a simulator.

OLB reviews the results of the annual evaluation examination. During the administration of written operator examinations, the examiner spotchecks annual written evaluation examinations given by the utility. Normally about six examinations are selected for spotchecks of examination adequacy and fairness of grading. Most recently at Three Mile Island, three RO and three SRO examinations were chosen representing high, medium, and low grades with about 3 to 4 hours being spent in the review. No written evaluation was given by the OLB examiner at the conclusion of the review. 53/ No further evaluation of the requalification program written examination is performed by the NRC.

I&E performs spotchecks to verify that the utility is carrying out its requalification program. This is a paperwork review which occurs at least once per year and is performed in the case of Three Mile Island by inspectors from the NRC Region I Office, King of Prussia. The persons who inspect training and qualification records specialize in that area.54/ Written reports are submitted to the utility.

Summary

The NRC's involvement in the administration of operator training is quite limited. Requirements in 10 CFR 50 and 10 CFR 55 are not comprehensive, and guidelines in ANSI standards are very limited in scope. The Operator Licensing Branch is quite limited in resources and is heavily overloaded. The OLB has not received the support which it has needed to carry out an effective job.55/ Too much reliance is placed on part-time examiners who lack the proper qualifications for licensing operators. There are no internal audits of the OLB to evaluate its effectiveness. The net effect is that NRC does not ensure a comprehensive level of knowledge of those persons who operate reactors or supervise their operation.

Babcock & Wilcox

Training Services Staffing and Organization

Although the utility is responsible for training its own operators it is impractical, if not impossible, for operators to receive the required operational and emergency training on the control panel of the generating facility. The cost of outages which would result from such training would be prohibitive. A simulator that accurately reproduces the operating characteristics of the facility involved and in which the arrangement of the instrumentation and controls of the simulator quite closely parallels that of the facility involved permits operators to receive the requisite training in an acceptable manner. Inasmuch as such simulators are quite expensive, most utilities do not possess them but, rather, utilize the simulator training provided by another organization.

The Babcock & Wilcox (B&W) Company operates a simulator, described earlier, for its customers. A series of courses built up around this simulator have also been discussed. Simulator training services that B&W provides to customers are exactly that. The B&W training department is a service organization which does what the customers ask for.56/

That B&W simulator training represents in terms of time only a small portion of the whole training program for reactor operators has been emphasized by B&W representatives. An operator could spend as little as 2 weeks every 2 years at Lynchburg. 57/ However, the emergency training which is covered in that short interval is of great importance to safe reactor plant operation, mitigation of the effects of casualties, and ultimately protecting the public. To state that "the utility is responsible for training" 58/ is, perhaps, an oversimplification. The B&W bears a great deal of responsibility for operator training.

To accomplish this task, the Training Services Section of the Customer Services Department is comprised of a manager, five senior instructors, two associate instructors, and other individuals responsible for simulator programming, seminar coordination, videotaping, and plant operation. The manager, an engineer, has been with the company for over seven years. None of the instructors is a degreed engineer, although all senior instructors were formerly licensed as senior reactor operators. Instructors are not required to periodically requalify as operators.59/

It is not clear how the Training Services Section is organized. The lead instructor stated that the manager of training services is not responsible to any degree for the formulation of the training program. Rather, the lead instructor said he was responsible for training specifics.60/

The manager stated that since taking over that position in 1972 he had instituted many changes to make the section more customer oriented and to staff the organization with experienced operators rather than engineers who lacked experience. The changes were based on consultations with the customer utilities in 1972-1974.61/

Management/Engineering Role in Training

The Training Services Section has essentially functioned independently of Nuclear Power Generating Division management, the NPGD Engineering Department, and Met Ed management. The head of the former Customer Services Department, of which the Training Services Section was a part until February 1979, stated that he never reviewed the content of the training program and that no such reviews were made from outside the Training Services Section. 62/ Regular meetings were not held between engineering and training personnel. 63/ In fact, there was a general lack of interaction and information flow between engineering and training. There was no formal mechanism to integrate design and training. 64/ One senior engineer was not sure if there was a training section.

That engineering and design personnel had little influence on training is not surprising, for engineers had almost no first-hand knowledge of nuclear reactor operations.65/ Neither the head of the

Engineering Department, a B&W engineer for over 20 years, nor the head of licensing, a B&W engineer for 25 years, had ever observed a B&W-designed nuclear steam supply system in operation at power. 66/ Nearly all engineers asked made similar responses. The head of the Design Section stated that he did not know if engineers from design had received training from Training Services. The head of the Engineering Department did not know if any engineer had ever observed courses given to the customers. He further estimated that less than 20 percent of B&W engineers had received simulator training. 67/ The vice president, NPGD, indicated that he had devoted little attention to training.68/

In a similar fashion, Met Ed management had not observed training conducted by Babcock & Wilcox for TMI operators. Lower-level management personnel from Met Ed had attended courses at B&W as a part of their own qualification or requalification, but no one had come to Lynchburg for the express purpose of evaluating the effectiveness of classroom and simulator training.69/

Training Contract

The mechanism by which training services are to be provided are defined in a contract drawn up between the Babcock & Wilcox Company and a purchaser which is generally a utility. In the case of Met Ed, a long-term training service contract was agreed upon in March 1975 and was to be effective until Dec. 31, 1979. This contract originally provided for three segments of training as follows:

- a base scope for annual operator requalification and replacement training;

- prescheduled, committed simulator instruction as may be mutually agreed; and

- other programs as may be mutually agreed.

Annual attachments to the contract provide rates for the base scope of effort as well as rates for prescheduled simulator training.

The contract was amended in August 1976 to redefine the base scope of effort and to specify the number of simulator hours and classroom hours which would be provided through 1979. A number of change orders have been issued to the original and amended contracts to include courses not covered by the base scope of effort. For example, in 1977 seven change orders were executed to cover cold license courses, startup certification courses, refresher training, chemistry courses, and video-tapes and instruction manuals, all in addition to the base scope of effort.70/

Training services are separately accounted for in the NPGD. On the basis of direct costs, this section has not lost money.71/

Course Administration

The three principal training courses as far as Met Ed has been concerned are cold licensing (New Plant Operator T301), hot licensing (Replacement Operator T303), and requalification (Simulator Requalification T304). These courses consist of both classroom and simulator instruction.

Review of records associated with classroom instruction in these three courses and interviews of training services personnel identified a number of deficiencies or non-optimal techniques which detract from the effectiveness of instruction. Included in these are the following:

- There is no indication that actual plant operating experiences are effectively incorporated in the classroom. For instance, the Davis-Besse transient of Sept. 24, 1977, was alleged to have been mentioned to some students from an unidentified utility, 72/ but there is no record of this nor is there an indication in Met Ed training records that such events from actual operations have been covered.

- There are no syllabi for these courses. Course content is determined when the weekly schedule is prepared.73/

- There are no training manuals used to standardize the material covered in the courses offered to TMI-2 operators and to assist in teaching. Documents such as the Final Safety Analysis Report and technical specifications are used, but these are not texts. A training manual was produced and used for TMI-1 operators. A similar manual was not produced for TMI-2 operator training because it was not in the contract.74/

- In the cold license and hot license courses, emphasis is given to teaching students to pass the NRC examination rather than concentrating on fundamentals. Students are given old NRC examinations to study.75/

- Weekly and final examinations are given in the cold license course. In neither the hot license course nor the requalification courses are quizzes given to test the effectiveness of classroom instruction.76/

- Homework is not assigned to students because bargaining units would require the payment of overtime. However, the training services manager believes that homework is not necessary because B&W provides training and not education.77/

- Lectures are not monitored and instructors are not monitored for effectiveness. Lecture evaluation sheets are prepared for and sent to the Three Mile Island Training Department. These evaluations are signed by a B&W instructor and are generally devoid of any comments.78/

- No audits of training methods and effectiveness are performed by anyone outside training services. Nobody in B&W management above the Training Services Section has given attention to course administration. 79/ There is little evidence that there has been a consciousness of the need for course upgrading at B&W.

Simulator Training

The B&W Training Center simulator, which is the so-called "SMUD" design,80/ adequately represents the controls and indication for TMI-2.81/ Prior to the accident on March 28, 1979, this was not true because the simulator was unable to simulate the formation of steam voids in the reactor coolant system. That is, it was not possible to simulate an increasing pressurizer level at the same time reactor coolant inventory was diminishing. This problem has been corrected since March 28, 1979, and the simulator can now represent a transient of the nature of that experienced at TMI-2. However, the simulator still lacks any indication of pilot-operated relief valve (PORV) position, either direct or indirect.

Although simulator training is considered by the operators to be valuable, the B&W simulator has not been employed adequately in light of the TMI-2 accident. Specifically:

- The evolutions and drills have been simple and repeated course after course. Compare, for example, the major drills covered in two simulator courses nearly 6 years apart:

June 1, 1973	March 23, 1979
Dropped Rod - 1	Dropped Rod - 2
Stuck Rod - 1	Stuck Rod - 1
Reactor Trip - 2	Reactor Trip - 2
RCS Leak - 2	RCS Leak - 1
Turbine Trip - 3	Turbine Trip - 2
	OTSG Tube Failure - 1
Steam Leak - 2	Steam Leak - 3

  Although the number of drills and evolutions conducted during a given period in the simulator has increased in recent years, the additional evolutions and drills have been minor in nature, involving, for example, instrumentation failures, valve malfunctions, or pump failures.

- As many as 10 to 15 casualties have been run in an hour. All would be single failure rather than multiple failure drills. Casualties were carried only through the immediate action steps rather than to their logical conclusion. For instance, a loss-of-coolant accident (LOCA) might not be carried past the point of emergency core cooling system (ECCS) actuation or getting a high pressure in the reactor building.82/ ECCS actuation was not simulated with subsequent failures.

- Casualties in the simulator were not dealt with by a crew; B&W discou-aged more than three persons in the simulator at a time, although more than this could be accommodated. There was no person who stood back and was in overall charge. All three trainees were involved in manipulations. The functions of extra reactor operators, the shift supervisor, or an engineer, were not simulated.83/

- Operators functioned based on their knowledge of procedures rather than breaking out and referring to the procedures themselves. This was, perhaps, due to the fact that the B&W Training Services Section did not have a set of TMI-2 operating and emergency procedures. 84/ Procedures for use in the simulator were available, but they were significantly different from those in use in the TMI-2 control room.

- Procedures that are required to be followed for a small-break LOCA and which were developed in response to an early 1978 B&W accident analysis were not used in the B&W simulator. These procedures require, among other things, for one reactor operator to be designated as a small-break LOCA operator and for this person to perform certain actions at prescribed times. The manager of training services was not aware that such assignments and actions were required.85/

- B&W training services has developed no drill guides or equivalent which would accomplish the following:

    - assign a specific objective for each drill;
    - provide a general description of the drill;
    - indicate the method of initiation, including the symptoms to be provided to the trainees;
    - provide a sequence of expected action;
    - specify the point of termination to ensure the objectives were achievable; or
    - permit management review and approval of the scope of training conducted.

- Trainee performance on the simulator is not evaluated in the requalification program; in the replacement operator program only the reactor startup is evaluated. This is particularly significant in the case of requalification because the simulator is normally the only practical casualty training that a reactor operator receives during the requalification cycle. There is no mechanism for reporting to the utility that a licensed operator did not perform satisfactorily in simulated reactor emergencies.86/

- Met Ed management has not observed the performance of TMI-2 operators in the simulator for the purpose of evaluating operator proficiency.87/

- One of the requirements of 10 CFR 55 is that each licensed operator perform at least 10 reactivity manipulations each 2 years. These manipulations may be performed on a simulator for the purpose of meeting requalification requirements. At the B&W simulator, all trainees present in the simulator have been given credit for a reactivity manipulation whether they operated the controls or not.88/ The NRC OLB chief was unaware of this practice and disapproved of it.89/

- The importance of keeping the core covered was not emphasized in the simulator.

- Natural circulation was not practiced to the point of cold shutdown.

Training Related to the Loss-of-Coolant Accident

The core of the TMI-2 reactor was damaged because operators failed to respond correctly to the symptoms of a loss-of-coolant through a stuck-open PORV. A complicating factor was pressurizer level increasing, rather than decreasing, at the same time that reactor coolant system pressure dropped. The operators did not recognize the symptoms of a LOCA, and they were incorrectly concerned about reducing the pressurizer level at a time when water should have been added rather than removed from the plant.

Inasmuch as the Babcock & Wilcox Company was the foremost authority concerning response of the B&W-designed plant to accidents and because it had a significant role in training TMI-2 operators to correctly react to accident symptoms, it is important that the training given to these operators with respect to a small-break LOCA be evaluated. The issues which relate to that training are:

- whether the B&W NPGD was aware that the plant would respond as it did on March 28, 1979;

- whether the manner in which the B&W plant would be expected to respond to a small-break LOCA from the pressurizer steam space was included in operator training and how operators were taught to respond;

- what B&W taught in simulator courses concerning permitting the pressurizer to go solid;

- what simulator students were taught about saturation conditions in the reactor coolant system; and

- whether operators were taught to keep the reactor core covered with water.

On Sept. 24, 1977, Toledo Edison's Davis-Besse-1 plant, which is similar to TMI-2, experienced a transient very similar to that which occurred in the March 28, 1979, event. The PORV for the pressurizer stuck open, causing rapid depressurization and actuation of the high pressure injection (HPI) system. Unexpectedly, a short time after the onset of the transient, pressurizer level began to rise. Relying on that indication alone, the operator terminated HPI. The transient continued for about 21 minutes until the operators shut the PORV block valve, terminating the loss of coolant, and stopping reactor coolant system depressurization.

The details of what occurred at B&W are covered in another section of the report. Aspects of those events which pertain to operator training will be mentioned briefly here.

An engineer from the B&W Plant Integration Unit was sent to Davis-Besse on Sept. 25, 1977, to review the transient. Upon his return, he briefed about 30 employees of his findings. On Nov. 1, 1977, because of his concern whether operators of B&W plants had been properly instructed about interrupting HPI following such an event, he wrote a memorandum 90/ in which he stated:

> Since there are accidents which require the continuous operation of the high pressure injection system, I wonder what guidance, if any, we should be giving to our customers on when they can safely shut the system down following an accident . . .? I would appreciate your thoughts on this subject.

The only written response which he received was a memorandum 91/ dated Nov. 10, 1977, from a supervisory engineer in Nuclear Service who pointed out:

> In talking with training personnel and in the opinion of this writer, the operators at Toledo responded in the correct manner considering how they had been trained and the reasons behind this training.
>
> My assumption and the training assumes first that RC (Reactor Coolant) Pressure and Pressurizer Level will trend in the same direction under a LOCA (loss-of-coolant accident). For a small leak, they keep the HP System up to a certain flow to maintain Presr. (Pressurizer) on Level.

He also stated that an instruction calling for continued operation of HPI might raise questions of vessel mechanics and of the RCS "going solid."

When no action had been taken on the matter, the manager of the emergency core cooling system (ECCS) unit wrote a memorandum 92/ dated Feb. 9, 1978, to the manager of licensing in which he stated

> The direct concern here rose out of the recent incident at Toledo (Davis-Besse). During the accident the operator terminated high pressure injection due to an apparent system recovery indicated by high level within the pressurizer. This action would have been acceptable only after the primary system had been in a subcooled state. Analysis of the data from the transient currently indicates that the system was in a two-phase (steam and water) state and as such did not contain sufficient capacity to allow high pressure injection termination. This became evident at some 20 to 30 minutes following termination of injection when the pressurizer level again collapsed and injection had to be reinitiated. During the 20 to 30 minutes following termination of injection when the pressurizer level again collapsed and 30 minutes of noninjection flow, they were continuously losing important fluid inventory even though the pressurizer indicated high level. I believe it fortunate that Toledo was at an extremely low power and extremely low burnup. <u>Had this event occurred in a reactor at full power with other than insignificant burnup, it is possible, perhaps probable, that core uncovery and possible fuel damage would have resulted.</u>
>
> The incident points out that we have not supplied information to reactor operators in the area of recovery from LOCA.
>
> I believe this is a very serious matter and deserves our prompt attention and correction. (emphasis supplied)

Action to inform reactor operators was not taken until April 4, 1979, a week after the TMI-2 accident. The guidance which was given to operators of B&W-designed power plants was essentially that contained in the Feb. 9, 1978, memorandum. Authors of both the Nov. 1, 1977, memorandum and the Feb. 9, 1978, memorandum agreed that if the operators had done what was contained in their memoranda there would have been no accident at Three Mile Island.93/

Aside from failing to inform the customers that HPI should not be interrupted during a loss-of-coolant accident, action was not taken within the B&W training organization to apprise operators of the details of the Davis-Besse transient or to discuss the implications of premature stopping of HPI. The manager of training service was at Davis-Besse on the day following the transient and was informed of the transient but did not understand its implications. <u>94</u>/ He was subsequently provided a copy of the Nov. 1, 1977, memorandum but cannot remember receiving a copy and took no action as a result. The Davis-Besse transient was not modeled in the simulator, and the lessons learned were not incorporated into the training program although an instructor may have discussed the transient with trainees from at least one utility.96/

Engineers occasionally discuss loss-of-coolant accidents in a 2-hour lecture on safety analysis to students undergoing requalification training. This was last done for Met Ed students on Jan. 23, 1978. The lecture has been given for the past 3 or 4 years, but not by the ECCS Unit; rather, it has been given by another unit of the Design Section, the Safety Analysis Unit, whose responsibilities did not include LOCAs. 97/ Similarly, LOCAs were not specifically covered in lectures given by training services instructors.98/

Considerable confusion exists about the issue of whether the pressurizer should be allowed to go solid, even among the key members of the Engineering Department. For example, the Plant Design Section engineer who investigated the Davis-Besse transient believes there is nothing wrong with going solid in the plant. 99/ The supervisory engineer in Plant Performance Services Section who wrote the Nov. 18, 1977, memorandum believes that operators are taught not to go solid at any time.100/ The manager of the ECCS Unit believes there is nothing written, "good or bad," about taking the plant solid.101/ The manager of the Plant Performance Services Section expressed concern in a memorandum dated Aug. 3, 1978, about allowing the reactor coolant system to go solid.102/

As far as training of operators at B&W is concerned, the manager of training services stated that trainees are not told not to go solid; in fact, they have not been taught anything specific about pressurizer level.103/ He was also not familiar with the B&W engineering department philosophy on going solid. Training services had not covered in the simulator the phenomenon of pressurizer level increasing at the same time reactor coolant system pressure was decreasing.104/ The operations manual for the B&W nuclear power plant simulator does not make any reference to pressurizer level while in the hot standby condition.105/

There are, however, requirements in various documents which clearly define what pressurizer levels should be maintained by the operators:

- Babcock & Wilcox Limits and Precautions106/ for pressurizer operations state:

  > The pressurizer must not be filled with water to indicated solid water conditions (400 inches) at any time, except as required for system hydrostatic tests.

- The pressurizer Limiting Condition for Operation in the Operating License Technical Specification107/ states:

  > 3.4.4 The pressurizer shall be OPERABLE with:
  >
  > a. A steam bubble; and
  >
  > b. A water volume between 240 and 1,330 cubic feet (45 and 385 inches).
  >
  > APPLICABILITY: Modes 1, 2, and 3.

- TMI-2 Operating Procedure OP 2103-1.3 Revision 3, 7/19/78, Pressurizer Operation, states:

-- 2.1.8 The pressurizer/RC System must not be filled with coolant to solid conditions (400 inches) at anytime except as required for system hydrostatic tests.

2.2.7 While in modes 1, 2 and 3, the pressurizer shall be OPERABLE with:

a. steam bubble; and

b. a water volume between 240 and 1,330 cubic feet (45 and 385 inches) (TS-3.4-4).

Neither the Limits and Precautions, Technical Specifications for Pressurizer Operation, nor the TMI-2 operating procedure were used in the training program at B&W.108/

Interviews and depositions of training services staff members as well as examination of training records indicate that the principle of keeping the core covered with water was not specifically treated in the training program. Additionally, the concept of saturation and translation of pressure-temperature relationships to saturation conditions were not emphasized. There was no steam table available for operator use in the simulator control room.109/

Courses Taken by TMI Personnel

Records of training conducted by Babcock & Wilcox for Met Ed operators since May 1973 were examined. Tabs A - H summarize these courses including type, length, dates, attendees, material covered, and results. Of particular interest is that drills involving a stuck-open PORV or loss of main and emergency feedwater were not covered in these 6 years.

TABS

A. Two-Week Simulator Course

B. Simulator Preparation Course

C. One-Week Special Simulator Course

D. Eight-Week Cold License Certification

E. One-Week Cold License Refresher

F. Special Five-Week Cold License Certification

G. Replacement Operator Training/Startup Certification

H. Requalification

Note: Those persons whose names are underlined are licensed either as a reactor operator or senior reactor operator on TMI-2.

TAB A

Two-Week Simulator Course

1. Completion Date	Attendees
May 18, 1973	Herbein3/, Porter, Smith
June 1, 1973	Floyd3/, Hydrick, Conrad
June 15, 1973	Colitz, Chwastyk, Miller
June 29, 1973	Gilbert, Potts, Smith
July 13, 1973	Beers, Desh, Ross3/
July 27, 1973	Kunder3/, Wallace, Fredland
Aug. 17, 1973	Guthrie, Noll, Evans, Book1/
Aug. 31, 1973	Derks, Boltz2/, Hitz, Mehler
Sept. 28, 1973	O'Hanlon, Baer, Hartman3/
Nov. 16, 1973	Bryan, Wynn, Heilman, Perks,4/ Goodlavage, Keisch, Morgan, Reich
Nov. 30, 1973	Banks, Acher, Zewe, Pilstz
Dec. 14, 1973	Bulmer, Keyser, Harper, Cotter
June 1, 1974	Brown2/, Crouse, Getty, Summers, Williams, Zechman2/
Feb. 14, 1975	Orlandi3/, Landers, Seelinger3/

1/ Now a member of B&W Training Services.

2/ Currently in TMI Training Department.

3/ Engineer/Manager.

4/ Former B&W Instructor.

5/ Persons whose names are underlined currently hold licenses on TMI-2.

2. Curriculum

	Hours
Simulator	40
Lecture	26
Study	10
Written Exam	4

3. Major Drills (Average number per person)

Reactor Trip	0.94
Turbine Trip	1.1
Steam Rupture	0.7
Dropped Rod	0.6
Stuck Rod	0.1
Dropped Rod	0.6
Stuck Rod	0.1
RCS Leak	0.7

4. None of the 54 course attendees failed.

5. Evaluations of the students were prepared. All lacked substantive comments.

TAB B

Simulator Preparation Course

1. Dates

April 23-26, 1973	17 Attendees
July 30 - Aug.3, 1973	20 Attendees

2. This special simulator preparation course which was of 4 days duration included lectures on fluids and electrical systems, components, plant startup and shutdown procedures, and some abnormal operations. A quiz was given at the conclusion of the course.

TAB C

One-Week Special Simulator Course

1. Completion Date / Attendees

Completion Date	Attendees
Oct. 18, 1974	Crouse, Banley, Brown2/
Feb. 3, 1978	Bezilla, Fuhrer
Feb. 10, 1978	Wearn, Fels
Feb. 17, 1978	Mackey, Crawford

2. Curriculum

	Hours
Simulator	20
Classroom	20

3. Major Drills (Average number per person)

Reactor Trip	.67
Turbine Trip	.22
Steam Rupture	.44
Dropped Rod	.67
Stuck Rod	.22
RCS Leak	.44

1/ Students prepared critiques.

2/ Currently in TMI Training Department.

TAB D

Eight-Week "Cold" License Certification

1. Inclusive Dates — Attendees

Jan. 10-March 4, 1978 — Thompson1/, Hartman2/, Faust, Coleman, Congdon

2. Curriculum

	Hours
Simulator	100
Classroom	180
Lectures/exams	141
Self-study	39

3. Major Drills (Average number per person)

Reactor Trip	2.0
Stuck Rod	0.20
Dropped Rod	1.4
RCS Leak	1.6
OTSG Tube Rupture	0.40
Steam Lean	2.6
Turbine Trip	0.40

4. Weekly and final examinations were given in each of the seven categories of the NRC written exam. In addition, startup, oral, and operating exams were given, all with satisfactory results.

1/ Due to lack of experience Thompson's oral was given at TMI 6/1/77.

2/ Employment terminated 4/79.

TAB E

One-Week Cold License Reference Course

1. Completion Date	Attendees
June 10, 1977	Beers,1/ Frederick, A. Miller, Congdon, Adams, Seelinger2/
June 17, 1977	Ross,2/ Hutchinson, Coleman, West, Desh, Hitz
June 24, 1977	Floyd,2/ Neumann, Hartman, Smith, Booher, Hoyt
July 1, 1977	Zewe, Marshall,2/ Wright, Mehler, Scheimann, Faust
July 8, 1977	Chwastyk, Illjes, Thompson, Guthrie, Tydon, Conaway

2. Curriculum

	Hours
Control Rod Drive	2
Diamond Panel Review	2
ICS Review	4
RPS	2
Safety Analysis	2
OTSG Review	2
Reactivity Changes	2
Review	2
Simulator	20

1/ Now in TMI Training Department.

2/ Engineer/Manager.

3. Major Drills (Average number per person)

Reactor Trip	.83
Turbine Trip	.47
RCS Leak	.43
Dropped Rod	.57
Stuck Rod	.17
OTSG Tube Failure	.60
Steam Leak	.83

4. There were no examinations, grades, or student evaluations in this course.

TAB F

Five-Week Cold License Certification Course (Special)

1. Inclusive Dates Jan. 16-Feb. 17, 1978

2. Attendee J. Logan, Prospective Superintendent, TMI-2.

3. Classroom

	Hours
Systems	36
Theory/Physics/Safety	16
Procedures	48
Exams	16
Tour	2
Review	6
Total	124

4. Simulator

	Hours
Planned Evaluation/Drills	66
Unannounced Drills	12

5. Major Drills

	Hours
Dropped Rod	6
Stuck Rod	2
Reactor Trip	5
RCS Leak	6
Steam Leak	4
Turbine Trip	1
OTSG Tube Rupture	1
PORV Stuck Open	0
Loss of Feed	0

6. Evaluation of Written Exams

Week one - Elementary level of knowledge required concerning plant materials air systems

Week two - Ten questions of the "list," "what is," "sketch," "describe" categories

Week three - Ten short answer questions concerning operational questions

Week four - Ten short answers on operational and emergency procedures required short answers to a specific problem or failure

Final examination - Fifteen short answer questions (10 of the questions were repeats of questions asked in the weekly study)

The documentation package includes 41 lesson/course attendance sheets and 41 lesson evaluation sheets. These are both Met Ed forms inasmuch as B&W has no requirement to keep attendance records or to evaluate lessons. The evaluations of instruction were all completed by H. Heilmer, a B&W instructor who himself presented 20 of 41 lessons. The evaluation sheets contain only brief statements of the material covered in the lessons and do not include any evaluation of the material covered, trainees performance, or instructor effectiveness, as they are required to do.

TAB G

T303-Replacement Operator Training/Startup Certification

1. Inclusive Dates / Attendees

Inclusive Dates	Attendees
May 23-June 1977	Marshall,1/ Tydon, Parnell
Oct. 31-Nov. 4, 1977	Wilkerson, Bailey
Oct. 31-Nov. 11, 1977	Masters
Nov. 7-11, 1977	Shipman1/, Goodman
March 20-31, 1978	Kidwell, Husted, Smith, Woodell
May 15-19, 1978	Cooper, Olson
June 26-July 7, 1978	Weaver, Bozer, Pearce
Oct. 2-13, 1978	McGovern, Logan,1/ Hemmila
Oct. 16-27, 1978	Zechman2/, Shipman1/, Hartman1/, Brantle'
Feb. 12-23, 1979	Kendig, Smith, Herman, Mayhue
Feb. 26-March 9, 1979	Phillipe, Garrison, Germer, Brumner, Mell

2. Curriculum

First Week	Second Week
Introduction (1)	Integrated Control System (2)
Control Rod Drive System (4)	Reactor Trip/Turbine Trip (2)
Nuclear Instrument/RPS (2)	ESF/RCS Leaks (2)
Reactor Physics (2)	RCP/MFP Trips (2)
Reactivity Balance (4)	Steam Leaks (2)
ECP Problems/Study (2)	Feed System Failures (2)
ECP Calculations (2)	CRD Malfunctions (2)
Oral Certification **Exam/** Study (2)	Instrument Failures (2)
	Review (4)

1/ Manager/Engineer.

2/ Supervisor of TMI Training Department.

3. Major Drills/Evaluations (Average number per person)

	Week One	Week Two
Reactor Trip	0.6	1.6
Turbine Trip	0.3	0.7
RCS Leak	0.3	0.5
Steam Leak	0.5	0.8
OTSG Tube Failure	0.16	0.15
Dropped Rod	0.5	0.5
Stuck Rod	0.0	0.2
Reactor Startups	6.9	5.6

4. Evaluation

All but two (Shipman, Goodman) of the 33 persons who took this course were administered a startup certification examination in accordance with Appendix F of the NRC Operator Licensing Guide. Copies of startup certification letters were sent to the chief of the Operator Licensing Branch, NRC. Review of the startup examination results indicate that none of the 31 persons being examined received an unsatisfactory evaluation in any category. Comments which were recorded are of little value in ascertaining operator performance.

5. Critique sheets were submitted by the students. They lack substantive comments and are replete with praise for the instructors.

TAB H

T304 Requalification (One Week)

1. Completion Date	Attendees
Jan. 13, 1978	Ross,!/ Parnell, Boyer, Chalecki, Floyd,1/ Brown,2/ Bryan
Jan. 20, 1978	Acher, Desh, Heilman, James, Seelinger,1/ B. Smith, D. Smith
Jan. 27, 1978	Crouse, Mehler, Kunderl/ Masters, Pilsitz, Zewe
Jan. 6, 1979	Acher, Heilman, Hartmanl/,
Jan. 12, 1979	Bryan, Parnell, Smith, Guthrie, Goodlavage, Brown,2/ Geruer
Jan. 19, 1979	Zewe, Brantley, Masters, Pilsitz, Mayhue, Ross,!/
Jan. 26, 1979	Chwastyk, Banks, Ruppert, Crouse, Hutchinson, Deiter, Flanagan
Feb. 5, 1979	Noll, Boyer, Keisch, Kandig, Mehler, Beers,2/ Husted
Feb. 9, 1979	James, Chalecki, Woodell, Hitz, Smith, Boltz,2/ Fraser
March 16, 1979	Congdon, Adams, Cooper, Mehler, Orwig
March 23, 1979	A. Miller, Wright, Olson, Coleman, Hitz, Marshall, Kunder
March 28, 1979 (Partial)	Floyd,1/ Smith
March 30, 1979	Hartman, Booher, Hoyt, Blessing

2. Classroom Curriculum

	Hours	
Push Pull	0-2	
Control Rod Drive Review	4	
Integrated Control System	4-6	
Heat Transfer	0-2	(Seldom included)
Power Distribution	0-2	
Heatup/Cooldown	0-2	
Safety Analysis	0-2	(Seldom included)
OTSG Review	0-2	
TOTAL	16-22	

3. Major Drills (Average number per person)

Reactor Trip	1.4
Turbine Trip	.55
Dropped Rod	.60
Stuck Rod	.32
RCS Leak	.55
Steam Leak	.96
OTSG Tube Failure	.40
PORV Stuck Open	0
Total Loss of Feedwater	0

Three Mile Island

Training at Three Mile Island for reactor operators, senior reactor operators, and auxiliary operators was significantly deficient. The training of operators did not prepare them to cope with the accident which occurred on March 28, 1979. Review of the events which took place indicate the persons in the control room should have been able to recognize symptoms which indicated the core was being hazarded and should have eliminated these hazards. However, that which was taught them did not ensure that they understood what was happening. Shortcomings in the organization, management, staffing, and curriculum as well as erroneous or incorrect training on specific issues related to the accident are discussed below.

TMI Training Department Staffing and Organization

The staff of the Training Department, as it existed on March 28, 1979, consisted of a supervisor of training, two technical training groups, and an administrative assistant. The technical training groups, one for licensed operator training and the other for nonlicensed operator training, were each headed by a group supervisor. The licensed operator training group included two instructors, and the nonlicensed group, one instructor.

The supervisor of the licensed operator technical training group is responsible for reactor operator and senior reactor operator initial training, administration of licensing requirements, and senior reactor operator upgrade training. The nonlicensed training group is involved in auxiliary operator training, health physics training, and ANSI requirements training for general employees.

The supervisor of training has been employed by Met Ed since 1969 when he was hired to organize a training department and a training program for TMI-1 operators. He does not have a degree and has been working unsuccessfully on a reactor operator license for more than 5 years. In September 1978, he became a full-time trainee for an operator license although he was not relieved of duties as supervisor of training. He was not able to closely oversee the training program because of his own qualification effort. 110/ The supervisor of training was less knowledgeable of training program particulars than members of his staff. For instance, he did not know, when asked, whether separate annual regualification written examinations are given to TMI-1 and TMI-2 operators. He was not familiar with the significant change in loss-of-coolant accident procedures which were instituted in May 1978. He was not aware of the procedure for examining license candidates with no reactor startup demonstration as described in Appendix F to the NRC Operator Licensing Guide; this is the procedure used by Met Ed for licensing replacement operators. He was not familiar with details of the annual oral examination procedure. ill/

The group supervisor of technical training for licensed operators has been at Three Mile Island for about 9 years. He is a qualified senior reactor operator on both TMI-1 and TMI-2 and has extensive experience as a shift supervisor. He does not have a baccalaureate degree.

The licensed operator group supervisor is knowledgeable of and involved in all aspects of operator licensing and training.

There are two instructors who train licensed operators. They both have extensive experience at TMI and both are qualified as senior reactor operators on TMI-1. Both are high school graduates.

The group supervisor of nonlicensed operator training does not hold either a reactor operator or senior reactor operator license. His single instructor is licensed as a reactor operator on TMI-1.

Working conditions for members of the training department have not been good. The many changes in the head of the training department have had adverse effects due to the lack of continuity. At times, the position has been vacant. At least five changes have taken place in as many years.112/ The training department has been undermanned, and the workload has been heavy. The group supervisor for training licensed operators says he has been overloaded since his first day in the training department. 113/ There has been a high turnover rate of operators since TMI-1 went commercial, which increased the magnitude of the training task; there have been as many as 20 persons in the Category IV (replacement operator) program. 114/ Another adversity has been that instructors are reportedly burdened by administrative requirements.115/

The lack of sufficient numbers of qualified persons probably has degraded the effectiveness of the training department.

Management and Training

Management's lack of involvement has also had a deleterious impact on the ability of the training department to function effectively. Neither the unit manager nor the station manager considered that he was responsible for the training of reactor operators nor had they been responsible for or involved in the content of the training program.116/ The training department was considered only to provide a service to use.117/

At the time of the TMI-2 accident, the training department did not, in fact, report to anyone on the Island. The supervisor of training was directly responsible to Met Ed's manager of quality assurance in Reading, Pa. In addition to being responsible for training of operators and quality assurance throughout the Met Ed system, the manager of quality assurance was also in charge of quality control and security. He stated that he visited Three Mile Island only about once per month and was able to devote little attention to training.118/

Training responsibility was moved to Reading during construction because of the inability to manage it at Three Mile Island. 119/ Responsibility was reassigned to the superintendent of TMI-1 in May 1979, but on June 26, 1979, neither the station manager nor the superintendent of TMI-2 were aware of this.120/

No in-depth audit of the functioning of the training department, no review of the adequacy of scope or depth of instruction given to operators, nor monitoring the effectiveness of simulator training at Babcock & Wilcox had been performed by Met Ed managers prior to the accident on March 28, 1979.

Evaluation of Training Programs

Training performed by the training department, as has been described above, consists of general employee training, auxiliary operator training, the replacement operator (Category IV) program, and requalification of licensed operators. General employee training takes place on an irregular, as-needed basis. Auxiliary operator "C" training is conducted on a full-time basis for 9 weeks before the operator carries out any duties in the plant. Auxiliary operators 'B' also spend 6 weeks, full time, in the classroom taking part in a formal curriculum. Training for auxiliary operators on shift is conducted in the same fashion as for licensed reactor operators and senior reactor operators; one week per shift cycle is spent in training. Prior to Jan. 1, 1979, there were five shifts, so operators spent one week out of five in training. Subsequently, the addition of shift 'F' meant that one week out of six was spent in the training department. The training week consists of the 7 a.m. to 3 p.m. shift, Monday to Friday.

Considerable reliance is placed on shift foremen and, to a lesser extent, on shift supervisors to train the persons in their shifts. But supervisors and foremen are normally heavily involved in administrative duties during routine shifts and are unable to devote attention to operator training. Estimates of the amount of time required for taking care of paperwork on shift vary between 60 to 75 percent. 121/

In addition to using solely training department assets, some use is made of consulting firms. For instance, the NUS Company of Rockville, Md., will perform nearly any training service needed. 122/ The training department has an NUS package that includes a modular program with video tapes.

General Employee Training. Training of general employees as recommended by ANSI 18.1-1971 is scheduled by department heads. This program will not be further evaluated here.

Auxiliary Operator Training. Auxiliary operator training is not defined formally in an administrative procedure. It is described informally in a training memorandum which was not approved by management. Review of the auxiliary operator (AO)/'C' training program indicates that the operators received adequate formal instruction to permit them to commence on-shift training. The material covered and the administration of the 9-week classroom program gives the 'C' AOs sufficient knowledge of systems and equipment. There is lacking, however, instruction of an integrative nature which would help them understand how the systems and equipment support the reactor. Additionally, it appears that auxiliary operators 'C' do not receive sufficient supervision to enable them to apply what they learned in the classroom to practice.

Auxiliary operator training outside the AO'C' and AO'B' classroom courses is ill-defined and sporadic. The goals of auxiliary operator training are not clear, particularly for A0'A's. Further, the training which has been performed has not enhanced their understanding of responsiblities with respect to reactor safety. Such responsibility is significant, as was emphasized by the events initiating the accident on March 28, 1979.

Examination of records for auxiliary operator training, other than the classroom sessions for AO'C's and AO'B's, indicates that between May 8, 1978, when formal training began for TMI-2 operators, and March 28, 1978, auxiliary operators went through 10 training weeks. Formal training during this time included the following topics:

- Industrial waste systems
- Portable survey equipment
- NET program
- Fire fighting
- General employee training -- emergency procedure
- Security review
- Evaporator training
- Caution tags
- Quality assurance
- First aid
- Radiological emergency drill
- Communications
- Administrative procedures
- Work request procedures
- Water and waste treatment
- Condensate polishers
- Snubber inspection

Of the 400 hours spent in the training department during these weeks, a total of 119 hours was devoted to formal training. During most of each week the training schedule indicated that auxiliary operators were not engaged in training. What they did during the approximate 70 percent of the time when they were not in formal training is not known.

Review of the topics covered shows that auxiliary operators received no training concerning reactor operation, reactor safety, radiological controls, radiation theory, chemistry, electrical safety, reactor theory, and so on. Such topics would be appropriate for operators who can directly effect reactor safety, who are exposed to radiation, and who are involved with radioactive material. The auxiliary operator training program requires significant upgrading to take advantage of the 40 hours which are available during each training week and to give emphasis to knowledge needed to ensure safe reactor operation and minimize radiation exposure to the auxiliary operators.

Replacement Operator Training (Category IV Program). The Category IV program, which over a 9-month period prepares candidates for reactor operator license examinations, is described by a training department administrative memorandum 123/ not approved formally by the supervisor of operations or any of his superiors. The program is completed essentially on a self study-basis. However, the training department does provide study assignments sheets which outline the trainees' required effort; reading assignment and physical tracing are listed for various systems and equipments. In addition, a questionnaire is provided for each system covered.

The program is fundamentally one of familiarization with the design, construction, operation, and interrelationships of systems and equipment in the plant. Included is coverage of administrative, operations, and emergency procedures. There is little emphasis given to theory, application of theoretical to the practical, or principles of either the operating or casualty procedures. Nothing is included in the course study assignments in the areas of reactor theory, primary plant thermodynamics, or secondary thermodynamics. There are no requirements for the study of such concepts as saturation, enthalpy, decay heat production, or solid system operation. Virtually no knowledge of chemistry is required. Rather, the emphasis is on systems, equipment, and procedures; topics included are listed in Table 1. Trainees may come off shift to attend lectures on specific topics such as reactor theory, integrated control system reviews, health physics review, or refueling reviews. Examination of weekly training schedules does not indicate that this is done in practice.

Review of questions included in the system questionnaires and in the examinations given to test the trainees' knowledge indicates that they are expected to memorize but not necessarily gain a comprehensive understanding of the material covered in the course curriculum. Questions are nearly all of the "what is," "list," "how many," "draw a one-line diagram of" type. There are few questions which begin with "why."

Trainees are expected to spend only about 2 hours per day on training. Checkouts to determine their level of knowledge are performed by the shift supervisor, shift foreman, or a licensed training coordinator. Checkouts are not performed by persons who might be expected to have a superior educational background such as the supervisor of operations or unit superintendent. There is no attempt to evaluate Category IV trainees' real depth of knowledge.

As a part of the replacement operation for qualification program, the trainees demonstrate their ability to conduct a reactor startup, change reactor power, and respond correctly to normal and emergency procedures at the B&W simulator. The TMI training department has little to do with the B&W startup certification. 124/ Met Ed managers do not observe the performances of trainees at Lynchburg.125/

At the conclusion of the Category IV program, a "mock" NRC examination is given before the NRC Operator Licensing Branch examiner comes to the site. Ideally, the mock exam would be given about 2 to 4 weeks before the actual NRC exam, but this period could be as long as 5 months.126/

The Met Ed replacement operator program is well administered. In addition, the program's success rate has been perfect. Of 43 persons applying to the NRC for reactor operator licenses since 1974, there have been no failures.127/ Nevertheless, the Category IV program does not ensure that reactor operators gain a broad understanding of the theoretical base for plant design or a comprehensive knowledge of the principles of the operating and emergency procedures.

TABLE 1: Topics Covered in Category IV Training Programs

<u>Cycle 1-1</u>

Administrative Procedures
Circulating Water Systems
River Water Systems

<u>Cycle 1-2</u>

Feedwater Systems Condensate Systems Main and Reheat Steam Systems
Extraction Steam/Heat Drains
Turbine Generator - Mechanical

<u>Cycle 2-1</u>

Make-up and Purification Systems
Decay Heat Removal Systems
Chemistry

<u>Cycle 2-2</u>

Safety Features Actuation System
Core Flood Systems
Reactor Building Spray Systems

<u>Cycle 3-1</u>

Turbine Generator Control Systems

<u>Cycle 3-2</u>

Reactor Protection Systems
Nonnuclear Instrumentation
Nuclear Instrument Systems
Reactor Coolant Systems
Reactor Coolant Pumps

<u>Cycle 5-1</u>

Health Physics
Radiation Monitoring Systems

<u>Cycle 5-2</u>

Reactivity Balance
Heat Balance

<u>Cycle 6-1</u>

Review:

- Administrative Procedures
- Operating Procedures
- Emergency Procedures
- Technical Specifications
- Reactor Theory

Senior Reactor Operator Training. Persons presently licensed as senior reactor operators at TMI-2 have gained this qualificaiton either through "cross licensing" or by means of the SRO "upgrade" program.

Cross licensing is a process by which persons who were qualified on TMI-1 obtained a license on TMI-2 without having to take an NRC license examination on TMI-2. SROs with TMI-1 licenses were administered a 10-week course in June-August 1978, which covered the differences in the design and procedure between TMI-1/TMI-2.128/ The NRC accepted documentation from the TMI training department that the candidates had participated in the differences program and had passed written examinations on these differences. The NRC reviewed the content of the differences examination. No oral examinations were given.

The SRO upgrade program is the method by which persons who are qualified as control room operators (CROs) prepare for NRC SRO license examination. This program too is not formally described in an administrative procedure. It is a total self-study effort, one month in length. The training department does provide study outlines and some guidance but this is essentially a self-help program. Success of candidates for SRO licenses has not been as good as for CRO licenses. Of 59 applicants for SRO licenses since 1974, 50 were successful.

Requalification. Met Ed's program for requalification of reactor operators and senior reactor operators is formally described and approved in Administrative Procedure (AP) 1006, previously described. Many deficiencies were noted in the content and administration of this program.

Operational Review Lecture Series -- (OR). The requalification program's principal means for ensuring operator knowledge on a continuing basis is the Operational Review (OR) Lecture Series. According to the requirements of AP 1006 the OR series will include the following topics as a minimum each year:

- reportable occurrences;
- unit modifications;
- operating history and problems;
- procedure changes;
- abnormal and emergency procedure review;
- technical specifications;
- major operational evaluations;
- applicable portions of 10 CFR; and
- FSR program material, that is, topics directly related to the different areas of the NRC licensing examination.

OR series lectures are required to consist of at least 60 scheduled hours per requalification cycle or about 1.2 hours per week.

Since the beginning of the requalification program in May 1978, operators were in five shifts until Jan. 1, 1979. Subsequently, they were in six shifts. The shift rotation is organized so that operators have one training shift per cycle. Thus, an operator would be expected to have a training week consisting of 40 hours, Monday to Friday, each 5 or 6 weeks.

Review of records indicates that both the amount of time devoted to training each training week as well as the material covered were deficient. In the 46 weeks between May 8, 1978, and March 28, 1979, the typical reactor operator or senior reactor operator was in training for nearly 10 weeks. During the 10 cycles between those dates (400 hours), the typical trainee received 214 hours of formal instruction. Whereas this exceeded the 60-hour minimum prescribed by AP 1006, it represented only about 53 percent of the time in which the operator was "in training." The remainder of the time was devoted to "operator in the plant" or nothing. "Operator in the plant" is ostensibly for the purpose of practical requalification work in the plant. In practice, however, it amounts to operators spending a shift in the plant just watching.

Some of the time supposedly devoted to lectures is misleading. For instance, the training schedule indicates a 6-hour period for emergency procedures. In actuality, 5 hours were for self-study on shift and one hour was for a one-hour test on emergency procedures. This test, which was given over a period of 5 weeks, was the only quiz given in the OR series; that is, no quizzes were given to determine the effectiveness of lectures. The shift supervisor of the March 28 shift (11:00 a.m. to 7:00 p.m.) failed this emergency procedure test which consisted of questions beginning as follows: "List," "How often," "What action," "What "List "List "List "List " "List " "List "What " "List," "How much," and "How long;" there was not a single "Why."

The material which was covered did not fulfill requirements of either AP 1006 or 10 CFR 55, Appendix A. Referring to the topics covered in the OR series after May 8, 1978, as listed in Table 2, no lecture was given which covered the details of reportable occurrences, their causes, and corrective action; operating history and experience in such a fashion as to cover significant lessons learned in other plants; in-depth discussion of procedure changes such as the modification to Emergency Procedure 2101-1.3, Reactor Coolant System Leak to incorporate small-break LOCA action; emergency procedure review (except for those procedures required to be covered under Health Physics Procedure 1670); or Fundamentals and System Review Program material.

As can be seen from Table 2 most of the lectures given were of little value in promoting safe reactor operation. The lectures which were related to reactor operation might be considered the following:

	Hours
Standardized technical specification review	4
Integrated control system	6
Reactor theory	10
Nuclear instruments	2
10 CFR	2
Operating history/experiences	1
Reactor protection system review	1
Reportable occurrences	2
Emergency procedures review	2
Reactivity balance	4
Features of facility design	2
General and specific operating characteristics	2
Instrumentation and controls	2
Safety and emergency systems	2
Decay heat removal system	8
	50

Thus, only slightly more than one hour per week was devoted to instruction related to reactor operation. Twelve hours of instruction on fire fighting, 129/ 4 hours on security, 8 hours of first aid training, 8 hours on fuel handling, and 32 hours of health physics/radiological emergency training were undoubtedly important. However, the lack of emphasis given to theoretical instruction concerning such material as decay heat, core cooling, thermodynamics, saturation, principles of operating procedures, and principles of emergency procedures was significant. The operators were not taught in such a fashion as to prepare them to cope with a situation not covered precisely in the emergency procedures.

Only one hour was devoted to "operating history/experiences." This lecture was the method by which operators would be exposed to transients and other phenomena experienced in other plants; its intent was to relate lessons learned to obviate repetition. Review of the lesson plan for this lecture indicates that the entire session was devoted to a discussion of relatively minor material problems, such as with reactor coolant pump snubbers, reactor building doors, and diesel engines. No personnel-related occurrences or events were discussed. In preparing for this lecture, the instructor reviewed the Licensee Event Report (LER) Summary dated Dec. 9, 1977, which included the transient at Davis-Besse on Sept. 24, 1977. The summary of that transient was vague and misleading in that it made no reference to the significant issues of the event.130/ The instructor did not understand the transient's significance.131/

Frequently, lectures were concentrated into 1 or 2 days a week as is indicated in Table 2. It is doubtful that covering such diverse topics as nuclear instruments, 10 CFR, operating history/experiences, caution tags, and quality assurance all in one session promoted effective learning.

Another significant deficiency in the requalification program was absenteeism from the OR series. Attendance at these lectures averaged less than 50 percent and was worse when the crew was on five shifts. 132/

TABLE 2: Reactor Operator/Senior Reactor Operator Formal Requalification Training. Mav 8. 1978 - March 28. 1979

May 8-June 9, 1978	Hours
Cross License Exam Review	2
Standardized Technical Specification Review	2
Integrated Control System (ICS) Review	2
Technical Specification Review	2
Operations/ICS Review	4
Emergency Procedure Review Techniques	1
Requalification Exam Review	3
	16

June 12 - July 14, 1978	Hours
Fire Fighting	8
General Employee Training-Emergency Procedures	2
Security Review	2
General Employee Training - Health Physics	4
	16

July 17-Aug. 18, 1978	Hours
Reactor Theory	8
Nucleax Instruments	2
10 CFR^	2
Operating History/Experiences	1
Caution Tags	1
Operational Quality Assurances	2
Fire System Technical Specifications	4
Radiological Emergency Drill	4
	24

Aug. 21-Sept..22, 1978	Hours
First Aid	8
Emergency Producures HPP 1670	--
Radiological Emergency Drill Training	8
	16

Same day

Note: D Shift got additional fire fighting training in lieu of reactor theory.

Sept. 24-Oct. 27, 1978	Hours
ISI Review *	2
Station Communications /Radiological Emergency Training *	2
Emergency Procedures HPP 1670 *	2
Reactor Protection System Review	1
Radiation Monitor System Review ^	
	15

Oct. 30-Nov. 19, 1978	Hours
Health Physics Recertification	4
Fuel Handling *	6
Underwater Video	2
Loose Parts Monitor _,.	2
Reportable Qccurrences~	2
Math Review	2
Radiation Monitor System	2
Emergency Procedures Review	2
Reactivity Balance	4
	26

Nov. 20-Dec. 2, 1978	Hours
Nothing	-0-

Dec. 4-11, 1978	Hours
Health Physics Review	4

Dec. 11-Dec. 22, 1978	Hours
Health Physics *Recertification	4
Reactor Theory *	4
Radiation Control	2
Fuel Handling	2
Features of Facility Design'	2
General and Specific Operating Characteristics*	2
Instrumentation and Controls*	2
Safety and Emergency Systems*	2
	20

Dec. 25-29, 1978	Hours
Nothing	-o-

* Same day

Jan. 1-Feb. 9, 1979	Hours
Circulating Water/River Water Systems	8
Administrative Procedures	1
Work Request Procedure	1
Security System Modifications	2
NET Program	4
Water and Waste Treatment	4
Industrial Waste Treatment	8
	28

Feb. 12-March 19, 1979	Hours
Decay Heat Removal System	8
Condensate Polisher	4
Turbine	8
	20

March 19-28, 1979

At Simulator

Appendix Z which consists of three memoranda submitted by the licensed-operator group supervisor indicates that the attendance rate for senior reactor operators was significantly lower than for control room operators. The memorandum, dated Feb. 16, 1979, also shows that the TMI-2 superintendent, TMI-1 superintendent, TMI-1 supervisor of operations, TMI-2 supervisor of operations, and TMI-2 operations engineer, all of whom were licensed operators, had attended no training in 1979.

Those persons who are absent from lectures are given "care packages" by the training department. These are make-up requirements which consist usually of study assignments related to the material covered in lectures that were missed. Persons who are absent are required to complete a form indicating they have made up the material and to return the form to the training department.133/ Low levels of attendance result in adding to the instructors' administrative burden.

The OR series, as well as other training conducted, is listed in a weekly training schedule prepared by the licensed operator group supervisor of training. There is no specific quarterly, semiannual, or annual plan on which the weekly schedules are based. The training department honors requests from management concerning particular training needs; otherwise, there is no input or review by managers.

Copies of weekly training schedules are submitted to unit superintendents and several managers at the company's Reading headquarters.

Lectures are neither evaluated for the purpose of upgrading nor are they monitored by managers as a method of ensuring high quality. 134/

On-the-Job Training (OJT). An important part of the requalification program is on-the-job (OJT) training. The intent of this is to ensure that operators participate in a minimum number of reactivity manipulations and take part in surveillance testing, equipment checkout, and equipment operation. There is a requirement that reactivity manipulations be diverse, that is, that many different types of evaluations be covered.

The administrative mechanism by which on-the-job training is recorded is the "OJT book" maintained in the control room. Operators are expected to enter their own evaluations and reactivity manipulations in the book. Appropriate supervisors are required by AP 1006 to review the participation of licensed personnel in the OJT program. In fact, this task is performed by members of the training department. 135/ Staff review of the OJT book indicates adequate numbers of reactivity manipulations with proper diversity were covered.

Shift supervisors and shift foremen who are assigned the responsibility for overseeing on-the-job training devote most of their attention not to monitoring reactor operation or training subordinates but to taking care of administrative requirements. SROs who are experienced as shift supervisors or shift foremen estimate that a foreman must spend from 50 to 80 percent of a shift on paperwork. Supervisors try to tour each unit each shift, but this is not always possible. 136/ Administrative requirements which restrict supervisory persons from overseeing training and observing the performance of plant operators are listed in Table 3.

Annual Evaluation Examination. A key part of the requalification program is the annual written evaluation examination which is required to ensure that CROs and SROs are maintaining an adequate level of knowledge. Although it is not required, Met Ed also requires an oral evaluation examination conducted in conjunction with the written test. Both of these examinations simulate the original licensing examination and the operating test conducted by NRC examiners.

The written annual evaluation examinations were in the format of NRC examinations; that is, the questions were arranged in categories A through G and H through L except that some categories were combined because of similarity. For example, category A on the requalification examination includes questions for NRC Category A (Principles of Reactor Operations) and Category H (Reactor Theory). Review of the questions asked indicates that the scope and depth of knowledge were approximately equivalent to those asked in NRC examination. 137/ Of particular interest was one question concerning the conditions which would warrant small-break LOCA response. This will be further discussed below.

According to the requirements of AP 1006, if a person receives less than 80 percent on any section of the examination he will attend the Fundamentals and System Review (FSR) program related to the failed sections. If a license holder scores below 80 percent on two or more sections of the annual written examination, he will be given an oral examination and be evaluated for an accelerated training program (relieved of all duties) or permitted to continue his duties and participate in an FSR program.

TABLE 3: Shift Foreman Administrative Requirement

1. Radiological Work Procedures
2. Work Requests
3. Key Log
4. Lifted Lead/Jumpers Log
5. Temporary Change Log
6. Special Operations Foreman
7. Batch Process Log
8. Transient Cycle Log
9. Callout Work Sheets (Overtime)
10. Vacation Book
11. Absentee Book
12. Reportable Occurrences
13. Document Review Surveillance
14. Out-of-Service Stickers (Including Weekly Audit)
15. Do Not Operate/Caution Tag Log (Including Weekly Audit)
16. Switching and Tagging Book
17. Fire System Removal From Service Log
18. Liquid Transfer Checklist Log
19. Individual Daily Time Report
20. Surveillance Reports
21. Secondary Logs
22. Out Building Logs
23. Control Room Logs (35-40 pages)
24. Shift Foreman Log
25. Operations Surveillance
26. Revision Review Book

Review of the results for the licensed operators who took the examination in February 1979 indicates that two persons received overall grades of less than 80 percent. One of these persons, a TMI-2 CRO, received a score of less than 80 percent in four sections and scored less than 70 percent in one section. This operator was dismissed a few days after the March 28 accident. The other person who failed overall was the TMI-2 supervisor of operations, who scored less than 80 percent in five of the eight sections, less than 70 percent in three of eight sections, and whose highest grade on any section was 83.9. This supervisor of operations had not, as of June 27, been given an oral examination, as required, nor had he participated in remedial work. Among TMI-2 CROs there were three operators who received grades of less than 80 percent on Section D (Instrumentation and Control) and three who scored less than 80 on Section C (General Operating Characteristics).

Annual oral examinations were administered to all licensed operators between January and March 1979. The purpose of the examinations was to determine an operator's knowledge of normal and emergency procedures, equipment operation, technical specifications, and emergency plans. The same checklist 138/ which specified systems and casualties to be discussed was used for all examinations. Review of completed checklists indicates that almost no individual item was unsatisfactory and nearly complete absence of critical comments. No one failed the oral examination.

Specific Training Deficiencies Related to the Accident

It is evident from a review of the events which took place on March 28 that operator error may have been a significant factor in the interpretation of information available and in the action which was taken or not taken in response to such interpretation. Such errors may have included the following:

- pressurizer level versus RCS pressure;
- recognition of LOCA;
- recognition of plant at saturation;
- recognition of need to remove heat or how; and
- recognition of significance of radiation levels or high temperatures/pressures in reactor building.

The purpose of this section is to evaluate these responses or failure to recognize conditions which existed in terms of the training that the operators had received at TMI.

<u>Pressurizer Level Versus Reactor Coolant System Pressure.</u> Interviews of operators and members of the training department yielded many statements to the effect that limits and precautions, operating procedures, and technical specifications forbid permitting the pressurizer level to go to indicated solid conditions while the plant is in hot standby. Instructors stated that they had taught operators not to allow the pressurizer to go dry and not to let it go solid because of the hazards of over

pressurization.139/ The pressurizer should not be allowed to go solid because of B&W limits and precautions. 140/ Reactor Coolant System Technical Specification 3.4.4 prohibits a pressurizer level of greater than 385 inches in Mode 3 even though nothing would probably be broken.141/ If the plant goes solid, the pressure could exceed 2,900 psig; if code safety valves did not lift at pressure of greater than 2,750 psig, again a Technical Specification Limit, could be reached.142/

All of these concerns are certainly valid in the nonemergency situation. What is not clear is what was actually taught to operators. Nothing was included in reviews of the emergency procedure for loss of coolant because the phenomenon of rising pressurizer level and decreasing system pressure had not been considered by the training staff. Pressurizer operation, which would include the limits and precautions of Operating Procedure 2103-1.3, was not covered in the requalification program. The matter of avoiding going solid was not covered at B&W simulator training. The Category IV training program (Cycle 4-1) did not include any discussion of pressurizer limits and precautions or hazards of going solid in other than Mode 1. No explicit training on not going solid in Mode 3 was referred to by the TMI training department. One deponent stated that not going solid had been stressed in training at TMI, but he did not indicate how or when.143/ The superintendent of TMI-1, who is licensed, an engineer, and when deposed was responsible for training at Three Mile Island, perhaps most clearly described the situation when asked whether operators had been instructed to keep an inventory in the pressurizer and to keep the pressurizer from going solid.144/

> I would say yes, but I don't think the instruction in either case, particularly with respect to the solid, was brought out in as clear cut a way as I have just answered the question. I would say that I have expressed a philosophy in answering the question but I could not go back to statements in training notes and dig out that specific statement out [sic] of the training notes from either B&W or Metropolitan Edison. I have expressed a philosophy that one keeps inventory in the pressurizer, and I have expressed a philosophy that one does not take the pressurizer solid.

The operators in interviews and depositions stated that the limits and precautions and technical specifications with respect to pressurizer level should not be violated. Nevertheless, it is evident that they also understood that although the existence of an emergency does not, *ipso facto,* give license to damage equipment, if the situation dictates, the operator may exceed the limits and precautions to move to a safer condition. 145/ This would suggest that the operators might not have recognized that an emergency existed when they lowered pressurizer level.

Failure to Recognize a Loss-of-Coolant Accident. There is evidence from the actions which were taken, as well as from statements made in depositions and interviews, that the operators did not realize a loss-of-coolant accident (LOCA) was in progress. 146/ For some time in the early stages of the accident operators believed that because of the absence of radiation alarms in the reactor building a steam line break must have existed rather than a loss of coolant. 147/ This was in spite of the existence of the following symptoms listed in Emergency Procedure 2203-1.3 indicating that a loss-of-reactor-coolant accident was in progress:

- rapid continuing decrease of reactor coolant pressure;
- high reactor building sump level;
- reactor building temperature alarms; and
- increasing radiation levels, beginning 19 minutes after the turbine trip.

Review of training related to the loss-of-coolant situation provides some insights as to why the operators may not have recognized that a small-break LOCA existed. The Loss-of-Reactor-Coolant/Reactor Coolant Pressure (Emergency Procedure 2202-1.3) was included in formal training only once during the year preceding the accident. 148/ The procedure was covered as a health physics rather than an operational requirement; Health Physics Procedure 1670.9 requires that EP 2202-1.3 be covered annually. The procedure was included in a 2-hour lecture along with four other emergency procedures. It is doubtful that the loss-of-coolant procedure was discussed in depth.

More significant in evaluating whether the operators were predisposed to act in a certain way or, more accurately, to not act in response to symptoms of a small-break LOCA may be revealed in a discussion of the small-break LOCA procedure itself.

On May 1, 1978, Babcock & Wilcox issued a document entitled "Analysis of Small Breaks in the Reactor Coolant Pump Discharge Piping for the B&W Lowered Loop 177 FA Plants," in which the worst-case small-break was determined to be at the reactor coolant pump discharge. Results of the analysis shows that it was necessary to use operator action during the early stages of the postulated small-break LOCA to effectively mitigate the accident consequences by achieving sufficient and balanced flow through all four high pressure injection (HPI) lines. Specifically, the B&W analysis suggested that the following actions be taken:

- Upon Emergency Safety Features Actuation System (ESFAS) signal, check for flow through both HPI trains.
- If no flow in one train:

  -- open pump header cross-connect valves;

  -- check HPI valve position and open if closed;

  -- secure flow through normal makeup line if flow is indicated; and

-- throttle HPI valves as required to balance flow and run out limits.

The analysis assumed, among other things, that no off-site power was available and that one complete train of high pressure injection failed. That is to say, even with these conditions, the core would be protected if the prescribed operator action were taken.

This analysis was evidently incorrectly interpreted by Met Ed. A letter from the vice president for generation to NRC dated May 5, 1978,149/ described the procedures change in response to the B&W analysis. Specifically, the letter stated:

> The control room LOCA operator will, within two (2) minutes of the event (small-break LOCA), analyze his indications and determine if there is a loss of offsite power cross connected with a diesel or make-up pump failure and a small break LOCA. In the event of that occurence . . . the Control Room LOCA operator will direct the make-up pump discharge cross connect valve opened and he will proceed to the HPI throttle valves in the auxiliary building and balance the flow between legs.

Thus, the bounding conditions for the B&W analysis, that is, loss of off-site power and failure of one HPI train, were interpreted by Met Ed to define when the action for a small-break LOCA would be taken. In other words, such action would only be taken in case of the extremely unlikely concurrent loss of off-site power and loss of one HPI train due to pump failure or diesel failure. The letter further stated that, "Each shift will be rebriefed at least once per month of the action required in the procedures."

On May 12, 1978, a change to EP 2202-1.3 was issued which added the following section:150/

2.2.2 Small-Break LOCA Response

2.2.2.1 Within 2 minutes of the LOCA the CR0 dedicated to recognition of a small-break LOCA 151/ must complete the following:

a. Verify that a small-break LOCA with single failure symptoms exists.

Symptoms: 1. SFAS initiation and only one make-up pump started, or
2. SFAS initiation and loss of the 2-2E

This indicates then that the interpretation of the May 5, 1978, letter to NRC was further misconstrued as a small-break LOCA because it was defined in terms of the loss of one HPI train or the loss of off-site power and the simultaneous loss of off-site power and one diesel, a most unlikely set of circumstances. This would possibly lead to the interpretation by operators that unless these circumstances were obtained there could not be a small-break LOCA.152/

Interviews of operators determined just that interpretation. The operators in the control room during the accident thought that to have a small-break LOCA it was necessary to have a loss of off-site power, a loss of a diesel or loss of make-up pump, and a loss of coolant.153/

Review of the annual written evaluation examination given to licensed operators in February 1979 indicates that the training department interpreted the situation in the same way as did the operators. Question F. 3(a) was: "List the conditions, as given in the emergency procedure, which would warrant small-break LOCA response." The correct answer to this question was: "(a) SFAS initiation and only one MUP started, or (b) SFAS initiation and loss of 2-1E or 2-2E."

It can be argued, therefore, that the operators failed to take action for a small-break LOCA on March 28 because the situation did meet the criteria, as defined in their training, for a small-break LOCA.154/

Another factor which may have influenced the operators' failure to realize the existence of a loss-of-coolant situation was that for 2.3 hours they did not recognize a stuck-open PORV. Their failure to do so has roots in the training which they had received. The emergency procedure review schedule subsequent to May 8, 1978, 155/ indicates that Emergency Procedure 2202-1.5, Pressurizer System Failure, which gives in Section B the symptoms for and action to be taken if pilot-operated (electromatic) relief valve (RC-R2) fails open, was not included.

One of the key symptoms of an open PORV is high temperatures (greater than 200°F) read on the discharge line. This temperature had been reading nearly 200° F before the accident because of relief or code safety valve seat leaks. During the time when the valve was stuck open, the operators repeatedly discounted discharge pipe temperature readings between 203°F and 283°F because they believed that temperatures about the same as those of the pressurizer or reactor coolant system (RCS), 553°F to 646°F, should have been indicated for a stuck-open valve. In fact, isenthalpic expansion of pressurizer water through the stuck-open valve should have resulted in discharge pipe temperatures ranging between 219°F and 302°F.156/ A fundamental knowledge of thermodynamics was not included in either the Category IV or requalification program nor was it expected by the NRC.

Most important, from the standpoint of training, of all the reasons for operators not recognizing the existence of a small-break LOCA was failure for the lessons learned in other transients such as at Oconee-1, Oct. 13, 1975, and at Davis-Besse-1 on Sept. 24, 1977, where PORVs stuck open, communicated to those persons who required that knowledge -- the operators. Serious concerns expressed by B&W engineers and by an NRC inspector who foresaw the significance of operator errors at Davis-Besse-1 and predicted serious consequences if the same error were repeated, were not made known to the utilities. As late as July 17, 1979, the group supervisor for licensed-operator training at TMI who was responsible for the content and conduct of such training had not heard of Dunn, Michelson, Cresswell, or Novak.

Failure to Recognize That the Plant Was At Saturation. In response to the high pressurizer level, operators throttled high pressure injection and increased let-down flow in a continuing attempt to reduce pressurizer level. This act resulted **in** reactor coolant system pressure dropping rapidly and almost steadily from normal operating pressure to about 1,000 psig. Pressure dropped slowly over the next 2 hours to about 600 psig. The operators interpreted this as a stable condition and were not concerned about the phenomenon. Relative stability of reactor coolant system pressure was the result, of course, of pressure decreasing to that corresponding to the bulk saturation temperature in the hottest part of the system. Difficulties with pressurizer heater operation and pressurizer temperatures equal to reactor coolant hot leg temperatures, which in turn were equal to surge line temperatures, did not indicate to the operators that the core was in jeopardy. There was apparent lack of understanding of and regard for the phenomenon of saturation.157/

Examination of Category IV and requalification training records does not indicate that either before or after licensing did operators receive instruction on the concept of saturation or translation of pressure/temperature relations to saturation. Training department staff stated that reactor coolant system thermodynamics were covered **in** the 9-month Category IV program 158/, but there is no indication of this in records associated with the program.

Lack of Understanding of the Need or How to Remove Heat. From their actions, neither operators nor supervisory persons who subsequently reported to the control room demonstrated concern for removing heat from the core or how this might be accomplished. There were indications that operators did not understand the approximate amount of heat being generated in the core from fission production decay and what the consequences might be of not removing this heat. There was no apparent concern for symptoms of core run covering and loss of effective heat removal capability. Increased nuclear instrument readings were interpreted as symptoms of positive reactivity. Hot leg temperatures increasing to off-scale values were dismissed as instrument malfunctions.

In addition to not comprehending the significance of symptoms of dangerous decay heat generation rates, the operators did not demonstrate knowledge of the methods available for heat removal. They did not realize that an open relief valve removes a great amount of heat from the system and shutting that valve without taking compensatory heat removal action will result in higher core temperature. There was no recognition that maintaining the secondary side of the steam generator at essentially the same pressure as in the primary side with both in saturation conditions will result in no heat flow from primary to secondary.

Again, there is no evidence that training included instruction on decay heat generation rates immediately following a trip and core thermodynamics in general. The training department thought core thermodynamics had been covered adequately in operators' previous (Navy) training. Instructors said that there was some coverage of this subject in the reactor protection systems (RPS) questionnaire in the Category IV program. Examination of the questionnaire does not support the contention. Core thermodynamics were not taught to the operators.

Failure to Recognize the Significance of Radiation Levels in Reactor Building. Increasing radiation levels in the reactor building observed as early as 19 minutes into the accident followed by numerous radiation alarms were not interpreted by operators as most unusual. Later, very high radiation alarms were not associated with probable damage to the core. Prompt action was not taken upon the receipt of radiation alarms to seal the source of radiation from the environment. In addition, the operators did not seem to understand the significance of increases in gross radiochemistry valves.

Review of the training program determines again that in-depth study of radiochemistry and quantitative evaluation of the amounts of radioactivity contained in the core of the reactor at power had not been done as part of training. There had been no exercises involving the calculation of the amount of radioactivity which would be released to the coolant from a fuel failure. Operators in training had not been required to demonstrate by means of calculation the airborne activity and direct radiation resulting from release into containment of normal coolant and coolant-containing fission products. They were not required to have a comprehensive knowledge of radiochemistry. No group problem-solving sessions had been conducted either with operators or health physics personnel to enhance their ability to estimate the likely causes of unusual radioactivity levels and measurements.

Summaries of Individual Qualifications

The training and qualification record of each licensed reactor operator and senior reactor operator at TMI-2 on March 28, 1979, was reviewed to determine whether the requirements of ANSI 18.1-1971, 10 CFR 55, and the Met Ed requalification program had been met. No significant discrepancies were noted. A summary is provided in Table 4. The qualifications of managers was also reviewed. These are summarized in Table 5.

TABLE 4: Operator Qualifications, TMI-2

A. CONTROL ROOM OPERATORS

NAME	YEARS EXPERIENCE	YEARS EDUCATION	DATE EMPLOYED	DATE AUX 'A'	LICENSE EXPIRATION DATE	LAST AT SIMULATOR
R. R. Booher	6	12	3/18/71	4/5/71	10/19/79	1/30/79
M. S. Coleman	6	13	1/4/74	1/4/74	10/19/79	3/23/79
J. R. Congdon	7	13	1/7/74	1/18/74	10/19/79	3/16/79
M. V. Cooper	7	14	10/4/76	10/4/76	7/5/80	3/16/79
C. C. Faust	7	13	12/5/73	12/5/73	10/20/79	7/8/77
E. R. Frederick	5	13	11/28/73	11/28/73	10/19/79	7/8/77
H. W. Hartman	6	13	1/28/74	1/28/79	Note 1	3/30/79
E. D. Hemmila	6	15	10/18/76	10/18/76	12/6/80	10/13/78
T. F. Illjes	7	14	2/9/71	4/5/71	10/19/79	7/8/77
J. M. Kidwell	7	12	3/12/74	3/12/74	6/23/80	3/31/78
H. A. McGovern	6	12	10/11/76	10/11/76	12/6/80	10/13/78
D. 1. Olson	8	12	3/20/71	4/5/71	6/27/80	5/19/78
L. 0. Wright	0	15	3/22/71	4/2/73	10/19/79	7/8/77

B. SHIFT FOREMEN (SRO TMI-2)

NAME	PREVIOUS EXPERIENCE YRS	EDUCATION YEARS	DATE EMPLOYED	DATE AUX 'A'	DATE CRO	DATE FOREMAN	EXPIRATION DATE	LICENSE LAST TIME SIMULATOR
C. D. Adams	8	12	10/6/75	NA	NA	10/6/75	10/19/75	3/16/79
W. T. Conaway	6	13	3/23/70	3/23/70	8/11/75	3/1/78	5/3/80	7/8/77
C. L. Guthrie	9	12	2/2/71	UNIC	UNIC	3/1/73	8/1/79	1/12/79
K. R. Hyot	10	12	4/2/71	4/5/71	8/11/75	5/9/77	10/19/79	3/30/79
A. W. Miller	0	16	4/4/73	4/4/73	8/11/75	8/1/78	9/1/80	3/23/79
F. J. Scheimann	8	12	3/5/73	3/5/73	8/11/75	2/13/78	5/3/80	7/8/77

TABLE 4 (Continued)

C. SHIFT SUPERVISORS (CROSS LICENSED SRO)

NAME	PREVIOUS EXPERIENCE YRS	EDUCATION YEARS	DATE EMPLOYED	DATE AUX 'A'	DATE CRO	DATE FOREMAN	DATE SUPERVISOR	LICENSE EXPIRATION DATE	LAST TIME SIMULATOR
K. P. Bryan	11	12	7/10/67		10/20/69	7/28/75	10/1/75	9/27/80	1/12/79
J. J. Chwastyk	Unknown	12	6/17/68					2/23/80	1/26/79
G. Hite	0	12	3/11/69		7/13/70	10/13/75	10/1/77	9/27/80	3/23/79
R. S. Hutchison	6	15	4/2/73	4/2/73	8/11/75	8/11/75	3/1/79	2/21/81	1/26/79
B. A. Mehler	2	12	5/9/67		10/20/69	8/23/76	4/1/78	10/19/79	2/5/79
B. G. Smith	9	12	10/27/58				9/1/74	8/2/80	1/20/79
W. H. Zewe	6	12	2/14/72			10/1/73	5/15/76	1/29/79	1/19/79

TABLE 5: Metropolitan Edison Managers' Qualifications

NAME	POSITION	NUCLEAR EXPERIENCE DEGREE(S)	YEARS	LICENSE
J. G. Herbein	V.P. Generation	BS	16	NO
L. L. Lawyer	Manager Generation Operations	BS	22	NO
G. P. Miller	Station Superintendent	BS	14	NO
J. L. Seelinger	TMI-1 Superintendent	BSMS	11	SRO
J. B. Logan	TMI-1 Superintendent	BS	20	SRO
R. W. Dubiel	Supervisor-Radiation Project Chemistry	BSMS	8	NO
M. J. Ross	Supervisor Operations TMI-1	NONE	14	SRO
J. R. Floyd	Supervisor Operations TMI-2	BS	Unknown	SRO
W. E. Potts	Superintendent Tech. Services TMI-1	BS	9	NO
G. A. Kunder	Superintendent Tech. Services TMI-2	BS	9	SRO
W. Marshall	Operations Engineer	BS	7	SRO
H. Shipman	Operations Engineer	BS	11	SRO
D. Berry	Operations Engineer	BS	15	NO
C. Seitz	Operations Engineer	BS	5	NO
T. L. Mulleavy	Radiation Protection Supervisor	NONE	18	NO
D. M. Shovlin	Superintendent of Maintenance	NONE	6	NO
R. E. Sieglitz	Supervisor of Main-tenance TMI-2	BS	12	NO

IV. FINDINGS

Analysis of the selection, training, qualification, licensing, and staffing of Three Mile Island operating personnel suggest the following findings:

1. There is no regulation concerning the minimum eligibility requirements for either reactor operators or senior reactor operators.

2. The NRC has not prescribed any training requirements for the qualification of operators.

3. The NRC has not prescribed any requirements concerning the education, experience, reliability, skill, stress fitness, psychological fitness, or criminal records of managers, supervisors, operators, technicians, or repair personnel of nuclear power plants.

4. The NRC has not prescribed any requirements concerning the experience levels of operators prior to their being licensed.

5. An operator of a nuclear power plant need not be a high school graduate.

6. No management personnel other than the operations manager require operator licenses.

7. The minimum required shift composition for operation of TMI-2 while the reactor is at power is one senior operator, two operators, and two nonlicensed operators. Only one operator need be in the control room.

8. The examining and licensing of operators is solely the responsibility of the chief of the Operator Licensing Branch of the NRC.

9. Regulations do not require a comprehensive level of knowledge of reactor operators or senior reactor operators.

10. A candidate for an operator's license need not actually conduct a reactor startup and shutdown to obtain the license. He need not demonstrate the ability to respond to emergency situations.

11. The program for training and qualification of auxiliary operators at TMI is not defined formally.

12. There is no formal program at TMI for training shift foremen or shift supervisors.

13. The Babcock & Wilcox training department does not have a formal program.

14. The NRC has no requirements concerning the qualifications of engineers and managers.

15. Auxiliary operators who can affect reactor power level and who handle radioactive material are not subject to any regulatory requirements.

16. The regulations do not address any aspects of the licensing process other than a written examination and operating test.

17. The NRC licensing process institutionalizes a shallow level of operator knowledge.

18. The NRC conducts a paper review of licensee training programs and a one-time-only review of simulator training programs.

19. The NRC has no formal criteria concerning licensee or B&W instructor qualifications.

20. The NRC does not conduct in-depth reviews of licensee or simulator training programs.

21. The NRC has no objection to the licensee or B&W teaching the NRC licensing exam; "mock" exams are encouraged.

22. Most examiners who prepare and administer operator license examinations do not themselves have reactor operating experience.

23. A person can fail several categories of the NRC operator licensing exam and still pass overall.

24. No candidate for a reactor operator license at TMI since 1974 has failed an NRC licensing examination; 88 percent of senior reactor operator candidates have passed on the first attempt.

25. The Operator Licensing Branch of the NRC is not audited by other parts of the NRC.

26. Once a person is licensed by the NRC he will not, except in rare cases, be again examined by the NRC as long as he participates in a company-administered requalification program.

27. The Operator Licensing Branch of the NRC in understaffed and overworked and has not been given the attention that is merited.

28. Babcock & Wilcox performs a crucial role in training operators for utilities which do not have a simulator.

29. B&W instructors are not required to requalify as operators.

30. The B&W Training Service section has functioned almost independently of both the B&W management and engineering as far as course content and conduct are concerned.

31. There is a lack of interaction between plant designers and training personnel at B&W.

32. Few senior engineers at B&W have any first-hand experience with nuclear reactor operations.

33. Met Ed management had not observed training of their operators at B&W.

34. Many deficencies exist in the administraion of courses at B&W such as not factoring into the program transients from operating plants, lack of syllabi, and lack of training manuals.

35. The B&W simulator was unable to reproduce the TMI-2 accident sequence prior to March 28, 1979.

36. Evaluations and drills conducted on the simulator at B&W have not trained operators to cope with major casualties.

37. Trainees' performance on the B&W simulator was not evaluated although this is the only opportunity available to determine if an operator is competent to function during a emergency.

38. Training at B&W did not instruct operators on how to deal with a small-break LOCA in the steam space of a pressurizer. This was the TMI-2 accident cause and had been the subject of much concern among B&W engineers following a similar transient at Davis-Besse-1 in Toledo, Ohio.

39. Babcock & Wilcox did not instruct trainees one way or another about allowing the pressurizer to go solid when the reactor is shut down.

40. Training which operators received at Three Mile Island did not prepare them to cope with the accident on March 28, 1979.

41. The TMI training department is understaffed in terms of quality and quantity. The supervisor of training has been unable to obtain an operator license in over 5 years of trying.

42. Management at Three Mile Island has not been involved in, nor has it considered itself responsible for, training of operators.

43. The training department reports to Met Ed headquarters in Reading, Pa., not to site management.

44. There have been many changes of head of training.

45. Shift foremen who are responsible for operator training are unable to give adequate attention to this task.

46. Auxiliary operator training is sporadic and ill defined and does not cover material needed by these persons to carry out their jobs.

47. Only about 30 percent of the time allotted to training weeks is used for training of auxiliary operators.

48. Replacement operator training is not formally approved and is done on a self-study basis.

49. The Three Mile Island operator requalification program is of low quality; the material covered is shallow, does not include topics required by 10 CFR 55, and is not related principally to ensuring safe reactor operation. Absenteeism is high.

50. The TMI-2 training program did not teach operators about:

    a. pressurizer level versus reactor coolant system pressure;

    b. recognition of saturation conditions;

    c. recognition of the need to remove decay heat and how to do it;

    d. recognition of the significance of high radiaction levels; or

    e. recognition of a loss-of-coolant accident.

ACRONYMS

AEC	Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	auxiliary operator
AP	administrative procedure
B&W	Babcock & Wilcox Company
CFR	Code of Federal Regulations
Davis-Besse	Davis-Besse Nuclear Generating Station, Toledo, Ohio
EP	emergency procedure
FSAR	Final Safety Analysis Review
FSR	Fundamentals and System Review Program
GAO	General Accounting Office
GPU	General Public Utilities Corporation
HP	health physics
HPI	high pressure injection
ICS	Integrated Control System
I&E	Office of Inspection and Enforcement (NRC)
LER	Licensee Event Report
LOCA	loss-of-coolant accident
Met Ed	Metropolitan Edison Company
NRC	Nuclear Regulatory Commission
Oconee	Oconee Nuclear Station, Clemson, South Carolina
OJT	on-the-job training
OP	operating procedure
OR	Operational Review Series
OTSG	once through steam generator
PORV	pilot-operated relief valve
RCS	reactor coolant system
RO	reactor operator
SER	safety evaluation report
SFAS	safety features actuation system
SRO	senior reactor operator

NOTES

1/ Para 50.54, Conditions of Licenses.

2/ Id.

3/ Id.

4/ Training and Selection of the Supervisor-Radiation Protection.

5/ GAO Report, EMD 79-67, p. 8.

6/ Deposition of N. Elliott pp. 21,25. Interviews of P. Collins 7/23/79.

7/ U.S. Nuclear Regulatory Commission Functional Organization (NUREG-0325), p. 34.

8/ 10 CFR 50.54, the Operating License; 10 CFR 50.34, the FSAR; and Appendix A to 10 CFR 55.

9/ This requirement is in addition to those contained in the regulations.

10/ NUREG¯0094, p. 5.

11/ See NUREG-0094, Appendix F.

12/ P. Collins, p. 50.

13/ P. Collins, p. 90. Union objected to concept of supervisory RO.

14/ 10 CFR 50.12.

15/ Appendix K.

16/ Completes Course T303, "Replacement Operator Training at Babcock and Wilcox."

17/ 10 CFR 55, Appendix A.

18/ A description of all B&W training courses is given in Appendix M.

19/ B&W Nuclear Training Services Catalog, Form T 301.

20/ See Appendix J.

21/ B&W Training Services Catalog, Form T303.

22/ B&W Training Services Catalog, Form T304.

23/ Appendix N.

24/ Interview with P. Collins, July 23, 1979. Mentioned also by N. Elliott (B&W), June 20, 1979, M. Beers (TMI), June 27, 1979.

25/ P. Collins interview, July 23, 1979, and desposition July 28, 1979 p. 57.

26/ Collins interview July 23, 1979.

27/ Collins deposition, July 28, 1979, p. 14.

28/ Ibid., p. 15.

29/ Appendix 0.

30/ Collins interview.

31/ Collins deposition, p. 9.

32/ Collins deposition, p. 18.

33/ I&E Procedure 41745B.

34/ Beers interview, June 27, 1979.

35/ Collins interview.

36/ Elliott deposition, July 3, 1979, p. 36.

37/ Ibid., pp. 40,41.

38/ Collins deposition, p. 47.

39/ Collins deposition, p. 39, and interview.

40/ Elliott deposition, p. 38.

41/ Collins interview, and deposition, p. 18.

42/ Collins interview.

43/ These groups are concerned with (a) Westinghouse; (b) General Electric, or (c) Combustion-Engineering, B&W, research reactors.

44/ Oak Ridge, Los Alamos, Livermore, Brookhaven, and Argonne Laboratories.

45/ Collins deposition, p. 42.

46/ Ibid., p. 44.

47/ Collins interview.

48/ Collins deposition, pp. 45-47.

49/ N. Brown interview, June 27, 1979.

50/ Collins interview.

51/ GAO Report B-127945, dated May 15, 1979, p. 12.

52/ Collins deposition, p. 25.

53/ N. Brown interview.

54/ Interviews of G. Miller, M. Beers, N. Brown, June 26, 1979.

55/ Collins deposition, p. 51.

56/ Elliott deposition, p. 135.

57/ Elliott Public Hearings, July 19, 1979.

58/ MacMillan Public Hearings, July 20, 1979.

59/ Elliott interview.

60/ Lind deposition, July 3, 1979, p. 15.

61/ Elliott deposition, pp. 7,8.

62/ Olds deposition, July 6, 1979, p. 23.

63/ Dunn deposition, July 2, 1979, pp. 29-31.

64/ Womack deposition, June 30, 1979, pp. 9-10, and Roy deposition, July 7, 1979, pp. 96-99.

65/ Willse deposition at 46 and 80.

66/ Roy deposition, p. 94, and Taylor deposition, July 6, 1979, p. 114.

67/ Roy deposition, p. 99.

68/ MacMillan Public Hearings, July 20, 1979.

69/ M. Beers deposition, p. 80, 81; Elliott interview, June 22, 1979.

70/ *See* Appendix P, Summary of Training Contract.

71/ Elliott deposition, p. 66.

72/ Elliott deposition.

73/ Elliott deposition, p. 31.

74/ Elliott deposition, p. 117.

75/ Elliott deposition, p. 161.

76/ The oral certification conducted at the end of the first week of course T301 is for other purposes.

77/ Elliott deposition, pp. 154-157.

78/ Appendix Q.

79/ Elliott deposition, p. 167.

80/ Named for the owner of the Rancho Seco nuclear power plant, the Sacramento Municipal Utilities District.

81 The simulator was observed in operation for several hours by members of the Commission staff with extensive nuclear reactor operating experience

82/ Zewe deposition, July 26, 1979, p. 39, and Seelinger deposition, Aug. 6, 1979, p. 181.

83/ Seelinger deposition, p. 189.

84/ Elliott deposition, p. 170.

85/ Elliott in Public Hearings, July 19, 1979.

86/ Elliott interview.

87/ Id.

88/ Elliott interview and deposition.

89/ Collins interview .

90/ Appendix R. A copy went to the Manager, Training Services.

91/ Appendix S.

92/ Appendix T.

93/ Kelly Public Hearing, July 18, 1979; Dunn Public Hearing, July 18, 1979.

94/ Elliott deposition, p. 70, et seq.

95/ Ibid., p. 75.

96/ Ibid., pp. 78, 79.

97/ Dunn deposition, pp. 25, 31.

98/ Elliott interview.

99/ Kelly Public Hearing, July 18, 1979.

100/ Walters Public Hearing, July 18, 1979.

101/ Dunn deposition, p. 228.

102/ Appendix U.

103/ Elliott interview.

104/ Elliott deposition, p. 143.

105/ Mode 3 or Hot Standby is with reactor shutdown and average coolant temperatures greater than 280°F.

106/ Appendix V.

107/ Appendix W.

108/ Elliott deposition, pp. 146-149.

109/ Elliott interview.

110/ Zechman deposition, p. 184.

111/ Zechman deposition, pp. 208-221.

112/ Brown deposition, pp. 35-50.

113/ Beers deposition, p. 87.

114/ Beer interview.

115/ Troffer interview, Brown interview, July 27, 1979.

116/ Miller, Logan interview, June 26, 1979; Miller deposition.

117/ Logan interview, June 26, 1979.

118/ Troffer interview, June 26, 1979.

119/ Miller interview, June 26, 1979.

120/ Miller, Logan interview, June 26, 1979.

121/ Interviews of Beers, Brown, and Scheiman, June 27, 1979.

122/ Beers deposition, pp. 28-29.

123/ Appendix K.

124/ Zechman interview.

125/ Beers interview.

126/ Id.

127/ Brown interview.

128/ Appendix X.

129/ Shift D received additional fire fighting training in lieu of 8 hours on reactor theory.

130/ Appendix Y.

131/ Brown interview.

132/ Beers interview.

133/ Brown deposition, p. 56.

134/ Beers interview.

135/ Brown interview.

136/ Beers interview, Scheiman interview, June 27, 1979.

137/ Appendix AA.

138/ Appendix BB.

139/ Beers interview.

140/ Id.

141/ Beers deposition, pp. 132-135.

142/ Beers deposition, p. 136.

143/ Zewe deposition, p. 93.

144/ Seelinger deposition, p. 200.

145/ Frederick deposition, p. 179.

146/ See Zewe deposition, p. 72, Frederick deposition.

147/ Frederick deposition.

148/ Between Sept. 25 - Oct. 28, 1978.

149/ Appendix CC.

150/ Appendix DD.

151/ It is unknown how a small-break LOCA would in fact be recognized. B&W intended the procedure for all LOCA's.

152/ The 2-1E and 2-2E boards normally are supplied by off-site power with diesel emergency supplies. Their loss implies loss of off-site power and one diesel.

153/ Frederick deposition, p. 214.

154/ Frederick deposition, p. 221.

155/ Appendix EE.

156/ See NSAC-1, "Analysis of Three Mile Island-Unit 2 Accident," Appendix ERV, Nuclear Safety Analysis Center, Palo Alto, Calif. 1979.

157/ Beers deposition, p. 71.

158/ Beers interview.

REFERENCES

1. General Accounting Office Report, EMD-79-67, May 15, 1979.

2. NUREG-0325, Functional Organization.

3. Nuclear Regulatory Commission, Inspection and Enforcement Manual.

4. NUREG-0094, Operator Licensing Guide.

5. Babcock & Wilcox Training Services Catalog.

6. Title 10, Code of Federal Regulations, Part 50.

7. Title 10, Code of Federal Regulations, Part 55.

8. Nuclear Regulatory Commission, Regulatory Guide 1.8.

9. American National Standards Institute Standard 18.1-1971.

10. American National Standards Institute Standard 3.1-1978.

11. Three Mile Island Nuclear Station Safety Evaluation Report.

12. Three Mile Island Nuclear Station Final Safety Analysis Report.

13. Three Mile Island Nuclear Station Operating License Technical Specifications.

13. Babcock & Wilcox, Limits and Precautions.

APPENDICES

A. 10 CFR 50, page 36.

B. 10 CFR 55.

C. Regulatory Guide 1.8.

D. ANSI Standard 18.1 - 1971.

E. SER Section 13.0, pp. 13-1, 13-2.

F. FSAR Section 6, Staff and Shift Organization.

G. FSAR Section 13.2.

H. Technical Specifications to Operating License Section 6.0.

I. Inspection and Enforcement Manual, Chapter 4100.

J. Appendix F to NUREG 0094.

K. Training Department Administrative Memorandum #5.

L. Administrative Procedure 1006.

M. Nuclear Training Services Catalog.

N. B&W Nuclear Training Center Diagram.

O. Appendix D to NUREG 0094, NRC Form 396.

P. Summary of Training Contract.

Q. Sample Lecture Evaluation.

R. J. J. Kelly Memorandum.

S. Walters Memorandum.

T. Dunn Memorandum.

U. Hallman Memorandum.

V. Babcock & Wilcox Pressurizer Limits and Precautions.

W. Pressurizer Limiting Condition for Operation.

X. Unit II Cross License Program.

Y. LER Summary dated Dec. 9, 1977, page 39.

---

*These documents are part of the Commission's permanent records that will be available in the National Archives under Accession #9290018.

Z. Memo from Training Group Supervisor on Absenteeism.

AA. Annual Evaluation Written Examination.

BB. Annual Evaluation Oral Examination.

CC. Letter to NRC Concerning Small-Break LOCA Procedure, dated May 5, 1978.

DD. Sample Simulator Evaluation.

EE. Sample I&E Training Review Report dated May 25, 1978.

FF. Category IV Program Questionnaire on Reactor Coolant System and Reactor Vessel Internals.

GG. NRC Form 157, Examination Report.

REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

TECHNICAL ASSESSMENT OF OPERATING, ABNORMAL, AND EMERGENCY PROCEDURES

BY

Ronald M. Eytchison

October 1979
Washington, D.C.

TABLE OF CONTENTS

SUMMARY

As a part of the effort to identify and evaluate the possible causes of the Three Mile Island (TMI) accident, an analysis of operating, abnormal, and emergency procedures was conducted by the staff. Of the 70 procedures included in these categories, 15 procedures were judged to be significant because they either were in use at the onset of the accident or became applicable as events took place. These procedures were evaluated for technical accuracy and adequacy with respect to the transient and its aftermath.

Evaluation of seven of the 15 procedures indicated that although they may be deficient in minor respects, they are adequate for their intended purpose. The procedures which are judged to be in this category are all operating procedures. They include: power operations, decay heat removal system, decay heat removal via once-through steam generator (OTSG), core flooding system, reactor building spray, emergency feedwater, and safety and safety features actuation system. The procedure for decay heat removal using the steam generators was considered to be clearly written and to provide a relatively simple, straightforward method for removing decay heat either with or without reactor coolant pumps in operation.

One operating procedure, one abnormal procedure, and two emergency procedures were believed to be usable, although they contain significant deficiencies that could cause confusion or lack of correct action:

- First, the reactor coolant pump operating procedure contains provisions that require tripping the pumps when vibration exceeds certain values. There is no discussion of unusual circumstances that might warrant pump operation with excessive vibration. Additionally, the vibration criteria conflict with those in another procedure. This procedure also does not provide clear instructions concerning whether the pumps should be tripped under low pressure, loss-of-coolant accident (LOCA) conditions.

- Abnormal procedure 2203-2.2 on turbine trip is deficient because it recognizes that the pilot-operated relief valve (PORV) will open on a turbine trip, but it does not include any precaution to ensure that the valve shuts. Also, the operator is required to let down coolant as necessary following a turbine trip to prevent the pressurizer level from exceeding 240 inches. This might contribute to operator action to avoid high pressurizer levels following a turbine trip, as occurred on March 28, 1979.

- The emergency procedure for loss of steam generator feed requires immediate tripping of the reactor following loss of both feedwater pumps, regardless of reactor power level. Additionally, the procedure does not recognize that the PORV would open, although such an occurrence would be very likely.

- Review of the reactor trip emergency procedure determined that its most significant shortcoming is lack of direction to determine the cause for the reactor trip.

The third categorization in this analysis identified one operating procedure, one abnormal procedure, and two emergency procedures that were evaluated as inadequate:

- The pressurizer operating procedure emphasizes that the operators are not permitted by the technical specifications -- a part of the operating license -- to exceed a pressurizer level of 385 inches in mode 3, the condition that the reactor plant was in following the reactor trip. There are no exceptions indicated in the procedure for emergency conditions. Thus, operators might be influenced by this procedure in their actions if a phenomenon not predicted by the procedure, such as rising pressurizer level following a reactor trip, were to take place. Although operators' actions should be governed by all of the symptoms available, this procedure is judged to be inadequate.

- The procedure for post-accident hydrogen control -- abnormal procedure 2203.2.6 -- fails to recognize that hydrogen can be generated rapidly, as occurred at TMI. It also does not recognize the difficulty of placing hydrogen recombiners into operation following an accident.

- Emergency procedure 2202.1.5 on pressurizer system failures is very confusing in its organization. Also, symptoms are significantly incomplete, misleading, or erroneous. Two sections of the procedure concerning a stuck-open PORV or stuck-open code safety valve should be in the LOCA procedure.

- From the standpoint of the TMI accident, perhaps the most important procedure was that for loss of reactor coolant/ reactor coolant system pressure -- emergency procedure 2202-1.3. The procedure does not provide the operators with objectives. It is difficult to use because the operator can be confused as to which section is applicable. A section on small-break LOCA response is misplaced, is illogical, and cannot be followed. The operator is required to bypass safeguards actuation and throttle high pressure injection, regardless of the severity of the accident. The procedure does not ensure that containment is isolated promptly.

Other general deficiencies were noted in the review of these procedures. Many minor substantive errors, typographical errors, and imprecise or sloppy terminology are not consistent with the quality required in nuclear power plant procedures. There also was noted a general emphasis on procedures to avoid equipment, component, or system damage and a lack of emphasis on keeping the core cooled.

INTRODUCTION

In accordance with the requirements of 10 CFR 50, Appendix B, commercial nuclear power plants are required to be operated as described in the operating and emergency procedures. Therefore, in considering the possible causes of the accident at Three Mile Island, investigation of procedures -- as well as personnel, design, and equipment factors -- was determined to be relevant. The technical accuracy and the adequacy of pertinent procedures deserved assessment to determine whether or not the procedures offered sufficient guidance for the control room operator.

TMI-2 has a total of 30 operating procedures, 25 emergency procedures, and 15 abnormal procedures. For the purposes of this analysis, it was decided that only those significant procedures that were in use at the onset of the accident or that were relevant as the accident progressed would be evaluated. Such procedures were determined to include nine operating procedures, two abnormal procedures, and four emergency procedures. Analysis of the procedures therefore was limited to study and evaluation of these 15 procedures.

ANALYSIS

The accident at Three Mile Island included a number of equipment, system, and overall plant events involving operating, abnormal, and emergency procedures. These procedures were intended to provide guidance to the operators concerning normal and abnormal plant operation, and if an emergency situation arose, the procedures were to direct mitigating actions to minimize the probability of plant damage and to ensure public safety.

The procedures that are evaluated in this paper according to their technical propriety and adequacy are listed below. They include significant procedures that were in use just before the onset of the accident -- procedures that were not referred to by the operators but that were pertinent.

- power operations -- OP2102-2.1
- pressurizer operation -- 0P2103-1.3
- reactor coolant pump operation -- OP2103-1.4
- decay heat removal system -- OP2104-1.3
- decay heat removal via once-through steam generator (OTSG) -- 0P2102-3.3
- core flooding system -- OP2104-1.1
- reactor building spray -- OP2104-1.4
- emergency feedwater -- OP2104-6.3
- safety features actuation system -- OP2105-1.3
- turbine trip -- AB2203-2.2
- post-accident hydrogen control -- AB2203-2.6
- loss of steam generator feed -- EP2202-2.2
- pressurizer system failure -- EP2202-1.5
- reactor trip -- EP2202-1.1
- loss of reactor coolant/reactor coolant system pressure -- EP2202-1.3

OPERATING PROCEDURE 2102-2.1, POWER OPERATIONS, REVISION 11, MARCH 20, 1979

The limits and precautions section of operating procedure (OP) 2102-2.1 states that in case a safety limit is exceeded (2.1), an automatic safety system does not function as required (2.2), or a limiting condition for operation (LCO) is not met, the shift supervisor shall notify the station/unit superintendent. It is not clear whether the procedure intends for the station manager, the unit superintendent, or both to be notified of such unsafe conditions.

The terms "operation" and "steady-state operation" are used but are not defined.

Limit 2.7, dated April 18, 1978, and section 4.1, dated June 17, 1977, state that the core thermal power shall not exceed 2,772 megawatts. In fact, core thermal power was restricted to 2,568 megawatts until late 1978, pending demonstration of full compliance with 10 CFR 50.46 and 10 CFR 50, Appendix K, concerning a small-break LOCA.

A prerequisite for reactor plant power operations, as listed in section 3.12, is that three independent steam generator auxiliary feedwater pumps and associated flow paths shall be operable. The term "operable," as defined in section 1.0 of the Three Mile Island Unit 2 (TMI-2) technical specifications, means that the system, subsystem, train, component, or device shall be capable of performing its specified functions.

Section 4.12 requires that shift logs be maintained in accordance with administrative procedure 1012.

Evaluation

The analysis of OP 2102-2.1 on power operations concludes that the procedure was adequate for the intended purpose.

OPERATING PROCEDURE 2103-1.3, PRESSURIZER OPERATIONS, REVISION 3, JULY 19, 1978

Paragraph 2.1.8 of the procedure's limits and precautions states:

> The pressurizer/RC [reactor coolant] systems must not be filled with coolant to solid conditions (400 inches) at any time except as required for system hydrostatic tests.

Paragraph 2.2.7 of the limits and precautions requires:

> While in modes 1, 2 and 3, the Pressurizer shall be operable with:
>
> a) Steam bubble, and
>
> b) A water volume between 240 and 1,330 cubic feet (45 and 385 inches).

These requirements are consonant with the Babcock & Wilcox Company (B&W) limits and precautions and with the operating license technical specifications.

Section 4.2.4 of the procedure provides instructions for equalizing pressurizer and reactor coolant system boron concentrations. This portion of the procedure would be used in case of pilot-operated relief valve or code safety valve seat leakage, such as was the case on the morning of March 28, 1979. Essentially, boron concentration equalization involves turning on heater bank 4 and opening the spray valve to permit spray flow into the pressurizer steam space.

<u>Evaluation</u>

Review of OP 2103-1.3, as well as pertinent portions of the technical specifications, emphasizes that the operators are required to avoid permitting the pressurizer level to exceed 385 inches in mode 3 -- the condition that the plant was in following reactor trip. Clearly, placing the core in jeopardy to avoid permitting the pressurizer from going solid is not the intent of either this procedure or the technical specification. However, there are no exceptions indicated in the procedure for emergency conditions. Thus, operators might be influenced in their actions by this procedure if a phenomenon not predicted by the procedure, such as rising pressurizer level following a reactor trip, were to take place. Although operators' actions should be governed by all of the symptoms available to them, this procedure is judged to be inadequate.

<u>OPERATING PROCEDURE 2103-1.4, REACTOR COOLANT PUMP OPERATION, REVISION 6, AUG. 16, 1979</u>

Of all the procedures reviewed, OP2103-1.4 has the most extensive list of limits and precautions -- about eight pages in length. Many of these are related to pump vibrations, including:

- The pump manufacturer shall be notified when reactor coolant pump steady-state vibration measured at the pump coupling reaches 15 mils peak to peak (section 2.1.1.15).

- Reactor coolant pumps must be tripped if motor stand vibration exceeds 3 mils (section 2.2.4.7), if shaft vibration of greater than 20 mils continues for 4 hours (section 2.2.4.7), or if shaft vibration exceeds 30 mils (section 2.2.4.8).

Note that abnormal procedure (AP) 2203-1.4, reactor coolant pump and motor emergencies and OP 2101-1.1 on nuclear plant limits and precautions both discuss shaft vibration and have limits that differ from those given in OP 2103-1 4.

The procedure requires, in section 4.3.2, that net positive suction head for the pumps be maintained in accordance with a curve provided in the procedure during low pressure condition in the reactor coolant system (RCS). Thus, at an RCS temperature of 582°F (nominal), the minimum allowed pressure for reactor coolant pump operation would be greater than 1,400 pounds per square inch guage (psig).

Evaluation

Operating procedure 2103-1.4 appears to be adequate for ensuring the proper operation of reactor coolant pumps. The conflicting criteria for what constitutes unacceptable pump or shaft vibration need to be resolved. Guidance concerning the pressure temperature relationship at which pumps must be tripped is quite clear.

The Nuclear Regulatory Commission (NRC) Investigative Report on the Three Mile Island accident (NUREG 0600) states categorically that reactor coolant pumps should have been tripped immediately when pressure dropped to 1,200 psig. This is in conflict with the statements contained in Inspection and Enforcement Bulletin 79-05A, which indicate that operators should not have tripped all reactor coolant pumps, even when flow had degraded significantly. It is suspected that the change in philosophy was the result of a B&W analysis after the accident, which determined that if high pressure injection initiates because of a low pressure condition in the reactor coolant system, all reactor coolant pumps should be tripped immediately.

OPERATING PROCEDURE 2104-1.3, DECAY HEAT REMOVAL SYSTEM, REVISION 11, JUNE 23, 1978

The decay heat removal system (DHRS) is designed to remove decay heat and sensible heat from the reactor coolant system during the latter stages of plant cooldown. In the event of a LOCA, the system injects borated water from the borated water storage tank (BWST) into the reactor vessel. For long-term emergency cooling, the system can take suction from the reactor building sump.

Operating procedure 2104-1.3 describes how the DHRS is used to fulfill these functions. Because the system is designed to operate at less than 340 psig, regardless of the RCS temperature, it must remain isolated from the RCS when system pressure and temperature are above prescribed values. The prerequisites of OP 2104-1.3 for placing the DHRS in operation state that the reactor coolant system should be cooled down to about 250°F and depressurized to less than 320 psig.

The DHRS normally is lined up for engineered safety actuation and will start when pressure drops below 1,650 psig or reactor building pressure increases to 4 psig. The system will operate in the recirculation mode, taking suction from either the BWST or the reactor building sump, until the reactor coolant system pressure drops to approximately 250 psig, at which time the system provides low pressure injection to the RCS.

This operating procedure also provides for long-term core circulation modes to prevent boron concentration effects after a loss-of-coolant accident. One of four long-term circulation modes should be placed into operation within 24 hours of the LOCA.

Evaluation

Operating procedure 2104-1.3 adequately describes how to operate the decay heat removal system for its intended purposes. The procedure was not used on March 28, 1979, because the required prerequisite conditions could not be achieved.

OPERATING PROCEDURE 2102-3.3, DECAY HEAT REMOVAL VIA OTSG, REVISION 6, APRIL 17, 1978

This procedure provides references, limits and precautions, prerequisites, and procedural steps for removing reactor decay heat using either reactor coolant pumps in operation or natural circulation cooling.

Prerequisites for placing the procedure in operation are listed in section 3.0 and are:

- The reactor is at "hot shutdown" (mode 3).

- The one-through steam generator (OTSG) level is being maintained at 97 to 99 percent in the operating range by means of the main or emergency feedwater pumps.

- Decay heat is being removed via the turbine bypass valves, with the turbine header pressure setpoint at 855 psig and maintaining the reactor coolant system temperature at 532°F.

To maintain reactor coolant system temperature or to cool the plant down with reactor coolant pumps in operation, it is only necessary to change the setting on the turbine header pressure set point to the desired value.

The procedure for removing decay heat by natural circulation cooling assumes the following initial conditions:

- reactor coolant pumps tripped, reactor tripped, or turbine tripped;

- steam pressure is being maintained at the turbine header set point (855 psig plus 125 psig following reactor trip), dumping steam to the main condenser through the turbine bypass valves or to the atmosphere if there is a low vacuum condition in the condenser;

- emergency feedwater pumps are maintaining OTSG level at 50 percent in the operating range; and

- pressure temperature limits are being maintained in accordance with figure 1.5.2. of the procedure.

Decay heat removal using natural circulation methods is accomplished simply by using the turbine header pressure setpoint to adjust OTSG pressure and thereby maintaining RCS temperature within the pressure/ temperature limitations of figure 1.5.2. This ensures adequate subcooling.

Evaluation

Operating procedure 2102-3.3 adequately describes procedures that will remove decay heat and either maintain a reactor coolant system temperature or cool the plant down to a desired temperature. The procedure is quite simple and straightforward. In fact, decay heat removal is accomplished with either forced or natural circulation by manipulation of only one control -- the turbine header pressure set point.

OPERATING PROCEDURE 2104-1.1, CORE FLOODING SYSTEM, REVISION 8, SEPT. 29, 1978

This procedure provides specific references, limits and precautions, prerequisites, and procedural steps covering system startup and shutdown.

Evaluation

The core flooding system is essentially a passive reservoir of water. This procedure adequately describes how the system is made ready for use and secured.

OPERATING PROCEDURE 2104-1.4, REACTOR BUILDING SPRAY, REVISION 3, APRIL 8, 1979

Operating procedure 2104-1.4 includes references, limits and precautions, prerequisites, and procedural steps covering operation of the reactor building spray system.

Evaluation

Operating procedure 2104-1.4 is adequate for the intended purpose of preparing the reactor building spray system for use and securing it.

OPERATING PROCEDURE 2104-6.3, EMERGENCY FEEDWATER, REVISION 4, JUNE 8, 1978

This procedure includes specific references, limits and precautions, prerequisites, and procedural steps for making the emergency feedwater system ready for operation. The procedure does not discuss operation of the system.

Two of the limits and precautions are of interest:

- 2.2.2 -- the emergency feedwater pumps will be put into standby during a unit heatup, after the first main feedwater pump has been placed in service, and EF-V11A and B have been placed in AUTO per 2102-1.1.

- 2.2.3 -- the maximum allowable number of cycles of the auxiliary feedwater nozzles in the OTSG is 80 for 80°F feedwater and 40 cycles for 40°F feedwater.

Emergency procedure 2202-1.3 includes a system valve lineup that specifies that emergency feedwater header isolation valves, EF-V12A and EF-V12B, should be open (see Appendix B).

Evaluation

The procedure appears to be adequate for making the emergency feedwater system ready for service, but it does not include any provisions for system operation.

The limit concerning the maximum number of times that cold water can be injected into the OTSGs would preclude conducting emergency feedwater pumps surveillance unless a valve were shut in the flow path to preclude flow into the steam generator.

Operating procedure 2104-6.3 is not rigorous in its terminology. Use of the terms "emergency feedwater pump," "emergency feed pump," "main feedwater pump," "main feed pump," "main feedwater (pump)," "emergency feedwater," "auxiliary feedwater," "main steam," "MS," "auxiliary steam," and "aux steam," in the same document could lead to confusion.

OPERATING PROCEDURE 2105-1.3, SAFETY FEATURES ACTUATION SYSTEM, REVISION 2, OCT. 25, 1978

This procedure includes references, limits and precautions, prerequisites, and procedural steps for system startup, normal operation, and shutdown.

Evaluation

Operating procedure 2105-1.3 is adequate to carry out the intended purposes.

ABNORMAL PROCEDURE 2202-2.2, TURBINE TRIP, REVISION 7, OCT. 25, 1978

Abnormal procedure 2202-2.2 describes the immediate automatic actions, immediate operator actions, and followup operator actions to be taken in case of a turbine trip.

Immediate automatic action step 2.OA.3 states, "Pressurizer Power Operated Relief Valve open . . ." indicating that PORV opening normally would be expected on a turbine trip. Step 2.OA.5 indicates that if both main feedwater pumps have tripped, the steam-driven emergency feedwater pump and two motor-driven emergency feedwater pumps will start.

Immediate operator actions following a turbine trip include:

° verify that the turbine stop valves are closed and generator and field breakers are open;

- verify the start of the seal oil backup pump, the turbine gear oil pump, and the bearing lift pumps and the closure of the extraction steam valves; and

- monitor pressurizer level and reactor coolant system pressure and temperature.

Followup operator action in section 3.0 includes:

- utilize pressurizer beaters and spray to control reactor coolant pressure at 2,155 psig and the steam header setpoint at 885 psig to control average coolant temperature at 582°F; adjust make-up and let-down flows to control pressurizer level at 240 inches; adjust feed flow to control once-through steam generator (OTSG) levels at 30 inches; and

- if the turbine trip is due to a loss of both feed pumps, verify that emergency feed pumps have started and are delivering water to the OTSGs; control EF-V11A and B to maintain OTSG levels at 30 inches.

Evaluation

Followup action in abnormal procedure 2203-2.2 does not include verifying the reclosure of the PORV, which apparently normally opens on a turbine trip. Otherwise, the procedure is adequate. Of interest is that the procedure recognizes that one of the likely causes of a turbine trip is loss of both running main feedwater pumps. In such an event, the procedure provides for making sure that emergency feedflow exists to both steam generators. The turbine trip procedure is written so as to minimize the amount of time that the plant is off the line. For instance, it prescribes that steam header pressure be maintained at 885 psig. The procedure also directs the operator to control pressurizer level at 240 inches, using let-down, if necessary. Whether this contributed to operator "mindset" -- do not let pressurizer level go high following a turbine trip -- cannot be determined.

ABNORMAL PROCEDURE 2203-2.6, POST-ACCIDENT HYDROGEN CONTROL, REVISION 1, JUNE 23, 1978

This procedure stipulates the taking of daily air samples for hydrogen in the reactor building following an accident and after the containment pressure reduces to normal and the activity level is reduced. Followup action calls for installing the hydrogen recombiner as the hydrogen concentration increases. If the hydrogen concentration cannot be maintained lower than 3.5 percent, purging of the reactor building must be commenced.

Evaluation

Abnormal procedure 2203-2.6 is seriously deficient in several respects:

- The term "accident" is not defined.

- No action is taken to even measure the hydrogen concentration until reactor building pressure has returned to normal and activity level is reduced.

- The procedure does not address the prompt generation of hydrogen in containment as it occurred at TMI. This appears to be a major oversight in view of the fact that a hydrogen burn actually took place on the first day of the accident.

- The procedure does not recognize the great difficulty in placing the hydrogen recombiner in operation because of radiation emanating from the recombiner and associated piping.

EMERGENCY PROCEDURE 2202-2.2, LOSS OF STEAM GENERATOR FEED, REVISION 3, OCT. 13, 1978

Emergency procedure 2202-2.2 lists symptoms, immediate actions, and followup actions for a loss of feedwater (FW) flow to both steam generators and a loss of flow to one steam generator.

Actions for a loss of both feed pumps include the following:

- 2.OA.1 -- Automatic Actions:

  -- If loss of FW is due to loss of both feed pumps:

    Reactor/turbine trip due to high RC pressure.

    Emergency feed pumps EF-P-l, EF-P-2A, and EF-P-2B start and maintain OTSG level at 30 inches (S/U range indication).

    If loss of FW is due to valves closing, ICS trips to track due to FW X-Limits.

- 2.OB.1 -- Manual Actions:

  -- If loss of FW is due to loss of both feed pumps:

    Trip the reactor.

    Verify turbine trip and stop valves closed.

    Verify EF-P-l, EF-P-2A, and EF-P-2B start as evidenced by pump discharge pressures.

    Verify emergency feedwater valves (EF-VIIA(B)) are on automatic and controlling level at 30 inches.

Evaluation

The procedure requires that the reactor should be tripped manually in case of any loss of both feedwater pumps, regardless of whether an automatic trip took place or not. This raises the question of why an automatic reactor trip circuit was not installed in the reactor protection system for a loss of both feedwater pumps.

It is not clear why verification of emergency feed pump operation and feedwater flow is listed under manual immediate actions rather than followup actions, or why there is a difference in this respect between the automatic and manual cases.

Emergency procedure 2202-2.2 does not make any mention of the almost certain operation of the PORV or checking to ensure that it functioned properly.

Otherwise, this procedure provides adequate guidance to the operator following loss of feedwater.

EMERGENCY PROCEDURE 2202-1.5, PRESSURIZER SYSTEM FAILURE, REVISION 3 SEPT. 29, 1978

Emergency procedure 2202-1.5 includes seven sections, each of which lists symptoms, immediate action, and followup action for the following casualties:

A. leaking pilot-operated (electromatic) relief valve (RC-R2)

B. inoperative pilot-operated (electromatic) relief valve (RC-R2)

C. leaking code relief valve (RC-R1A or RC-R1B)

D. inoperative code relief valve

E. inoperative pressurizer heaters

F. malfunction in pressurizer level indication or control

G. pressurizer spray valve failure

At some time during the accident on March 28, 1979, actual or suspected conditions applicable to sections A, B, C, E, and F existed.

Evaluation

Analysis of EP 2202-1.5 points out a number of deficiencies:

° The procedure is hard to use because there is no introductory section that indicates the contents or scope. Although the procedure is entitled "pressurizer system failure," it includes a variety of problems that might not be associated with a failure of the pressurizer system. There also should be an index.

- Section A (Leaking PORV)

  Symptoms do not indicate that one cannot tell whether high relief valve discharge line temperatures are due to the leaking of a PORV or a code safety valve.

  The symptoms suggest incorrectly that there is no indication of reactor coolant drain tank (RCDT) temperature in the control room.

  The immediate action section requires the PORV isolation valve (RC-V2) to be shut any time the discharge line temperature exceeds 130°F.

  There is no mention in followup action of the need to recirculate water through the pressurizer to equalize boron concentrations.

- Section B (Inoperative PORV)

  -- The symptoms for a stuck-open PORV do not include a discussion of position indication. The procedure assumes that this will be noted by the operator.

  -- The symptoms for a stuck-open PORV do not mention: increasing RCDT level, rapidly decreasing make-up tank level, possible increasing pressurizer level, or decreasing reactor coolant system (RCS) pressure.

  -- Possible increasing reactor building temperature, pressure and sump level.

  -- Automatic action indicates that all pressurizer heaters will be on below 2,105 psig. No mention is made about the possible loss of pressurizer heaters, which reportedly has been a chronic problem and was severe during the TMI accident.

  -- The action section does not include a warning against premature interruption of high pressure injection (HPI).

  -- The procedure requires the PORV block valve to be shut if the PORV itself fails shut.

  -- The entire section on a stuck-open PORV should be in the loss-of-coolant emergency procedure. This was precisely the cause of the loss-of-coolant accident at Three Mile Island.

- Section C (Leaking Code Relief Valve)

  -- The term "code relief" valve rather than the proper term "code safety" valve is used throughout. This improper terminology could cause confusion.

-- The same comment made above for a leaking PORV can be made here. Symptoms indicate erroneously that you can identify whether the leaking valve is the PORV or a code relief (safety) valve without taking any further action.

-- Followup action directs that a reactor coolant leakage rate measurement be taken. However, no mention is made of performing subsequent leakage rate measurements to determine trends.

• Section D (Inoperative Code Relief Valve)

-- The first two symptoms -- code relief fails to open, code relief fails to close -- are causes rather than symptoms.

-- Symptoms do not include RCS pressure dropping, increasing RCDT level, rapidly decreasing make-up tank level, possible increasing pressurizer level, and possible increasing reactor building temperature, pressure, or sump level.

-- For a fail-to-open code relief, the procedure does not mention as automatic action the de-energization of pressurizer heaters as it does in section B.

-- The term "safety injection" is used rather than "high pressure injection" as in section B.

In followup action, the procedure cautions against inserting any positive reactivity. With the reactor already fully shut down, it is not clear how positive reactivity would be inserted or what difference it would make.

-- The terms "code safety" and "code relief" are used interchangeably.

• Section E (Inoperative Pressurizer Heaters)

-- Followup action requires that if pressure cannot be maintained with the remaining heaters, continue load reduction to shutdown and possibly to a cooldown condition. This directive is not clear cut and positive. It should state exactly the criteria to determine if the plant can stay in hot shutdown or if it must be placed in cold shutdown.

• Section F (Malfunction in Pressurizer Level Indication or Control)

Symptom F.1.2 states, "Rapid change is indicated/ recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction." This statement, coupled with the title of the section, suggests that the procedure is applicable to the case in which there is an actual pressurizer level control malfunction as opposed to a level indication malfunction. However, the procedure action paragraphs provide no guidance for actual pressurizer level control problems.

EMERGENCY PROCEDURE 2202-1.1, REACTOR TRIP, REVISION 6, OCT. 25, 1978

This procedure provides symptoms and immediate and followup action steps for a reactor trip. Action required includes the following:

- Manually trip the reactor.
- Verify all in-limit lights are actuated, with the exception of group 8 rods.
- Close let-down valve MU-V376.
- Start a second make-up pump.
- Open MU-V16B as necessary to maintain 100 inches in the pressurizer.
- Verify that pressurizer heaters are off at 80 inches in the pressurizer.
- Announce "reactor trip" on page system.
- Monitor make-up tank level and maintain a level higher than 55 inches.
- Verify that pressurizer heaters and spray have returned RCS pressure to normal operating pressure (2,155 psig).
- Reduce pressurizer level set point to 100 inches.
- Verify that turbine bypass control valves are maintaining header pressure at 1,010 psig.
- Verify normal electrical lineup.
- Check that all RMS channels are normal.
- If reactor startup is not intended within 4 hours, raise OTSG level to 97 to 99 percent in the operating range.

Evaluation

Emergency procedure 2202-1.1 on reactor trip, in conjunction with abnormal procedure 2203-2.2 (turbine trip), provides adequate guidance to operators for this casualty. The obvious intent of the procedure is to place or verify the plant in a safe condition and to minimize the length of the outage.

A major shortcoming of the procedure is that it does not mention determining the cause of the reactor trip and correcting it. For example, the trip could have been caused by high RCS pressure, low RCS pressure, high reactor coolant outlet temperature, or high reactor building pressure. Each of these conditions could be the result of a hazardous set of circumstances that would have to be dealt with to stabilize the plant.

EMERGENCY PROCEDURE 2202-1.3, LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM PRESSURE, REVISION 11, OCT. 6, 1978

Background

The most serious accident that can occur in a pressurized water power reactor is a loss-of-coolant accident (LOCA) or loss-of-coolant pressure. Either could result in core damage due to the lack of heat removal. Emergency procedure 2202-1.3 is intended to ensure that in case of the most severe LOCA or loss-of-coolant pressure that systems and operators will function to prevent core meltdown, to maintain the integrity of the containment, and to ensure that the public is not exposed to radiation in excess of 10 CFR 100 limits. Similarly, the procedure should mitigate any LOCA or loss-of-coolant pressure of lesser severity.

Emergency procedure 2202-1.3 has as its analytical basis the Final Safety Analysis Report (FSP), Section 6. The FSAR assumes that the worst-case break is a 5 ft break in a reactor coolant system hot leg. Cold-leg breaks of various sizes are analyzed but are not considered as severe because less energy is released for a given size break.

A LOCA/loss-of-pressure event could be described as falling into one of the following three categories of severity, and the procedure must provide for each one:

1. A Leak. Mass loss from a leak can be accommodated from normal pressurizer water inventory and normal system pressure control. No safety system operation is necessary. Plant shutdown might be required due to technical specification limits on leakage, but the shutdown would be deliberate and orderly using normal procedures.

2. Small-Break LOCA. Engineered safeguards systems automatically activate and deliver water to the reactor coolant system. The break size is such that reactor cuilant system pressure is above the shutoff head of the low pressure injection pumps. All injection water is delivered by the high pressure injection

systems. Decay heat is removed by fluid exiting the break, by heatup of high pressure injection water, by heatup of the auxiliary feedwater delivered to the steam generators, and possibly by steam release from the steam generators. Containment isolation upon pressure rise in containment may not occur or may be delayed significantly.

3. Large-Break LOCA. Engineered safeguards systems automatically activate, as does the containment isolation signal from containment building pressure. The low pressure injection system delivers significant mass flow to the reactor coolant systems. Core flood tanks discharge to the reactor coolant system. Steam generators do not remove heat.

Evaluation

General Comments

The procedure does not identify an objective. The ultimate goal is not discussed. Symptoms are not listed in any priority, either by importance or by likelihood of occurrence. There is no statement to the effect that not all symptoms need be observed for the event to be taking place. The format of the procedure is difficult to follow; this has been exacerbated by the illogical insertion of a small-break LOCA subprocedure. Terminology is confusing. For instance, the following terms are used interchangeably to express the same phenomenon: high pressure injection, HPI, safety injection, ESF (engineered safety features).

The procedure has no guide to indicate what is contained inside; there is no list in the beginning of cases (similar sets of conditions which would dictate similar courses of action).

The procedure does not list any initial conditions or applicability such as reactor critical; reactor at power; reactor in hot standby; modes 1, 2, 3, 4, and so on.

Specific Comments

The first category of the procedure is "Leak within Capability of System Operation." This is confusing because it does not specify what system is capable of operating -- make-up system, high pressure injection system, reactor coolant system, or nuclear steam supply system.

° Symptoms

-- Symptom 1.1 is described as "Initial loss of reactor coolant pressure and decrease in pressurizer level becoming stable after a short period of time." This symptom <u>assumes</u> that correct action has been taken to correct the loss of pressure and level.

-- The symptoms do not include reactor coolant drain tank pressure or temperature increases increasing reactor building pressure or reactor coolant system leakage calculations.

- Manual action makes no provision for tripping the reactor on low reactor coolant system pressure even if the pressurizer level is not yet low.

- Manual initiation of high pressure injection does not mention the small-break LOCA response which is appropriate in this section.

- If high pressure injection is initiated manually, the operator is required to throttle HPI as necessary to maintain pressurizer level at 220 inches (that is, not permit it to increase).

- Step 3.2.8.1 says, "THROTTLE HPI string(s) flow rate to at least 500 gpm each 250 gpm per leg)." This is confusing.

- The procedure does not mention ensuring that containment isolation is set. The last statement in section A following a three-page discussion of how to place the decay heat removal system in operation **is**, "Reactor Building Isolation Initiated." It is not clear whether this is merely a statement or a much belated procedural step.

- Section B of the procedure is entitled "Leak or Rupture of Significant Size Such that Engineered Safety Features Systems are Automatically Initiated." It is not clear that this includes the procedure for a small-break LOCA.

- Symptoms do not:

  recognize that pressurizer level may increase, as it did on March 28, 1979;

  mention a high make-up flow alarm;

  -- mention reactor coolant drain tank high temperature or pressure; or

  -- mention a possible decrease in reactor coolant system flow.

- The procedure provides symptoms that are intended to determine whether a reactor coolant system leak, a steam leak, or an OTSG tube rupture are giving the observed symptoms. It is not clear why a steam leak would result in high pressure injection. Symptoms of an OTSG tube failure do not include increasing steam generator water level or pressure.

- The procedure states in one place (B.1.1.3) that safety injection commences at 1,640 psig and in another place (B.2.1.3) at 1,600 psig.

- Under manual action is a section on small-break LOCA response:

  -- This section is misplaced.

  -- The operator is required to verify that a small-break LOCA exists. How this is to be determined is not indicated.

  -- The section has its own illogical symptoms (of a small-break LOCA with single failure), which are:

    Safety features actuation system (SFAS) initiation and only one make-up pump (MUP) started; or

    SFAS initiation and loss of 2-1E or 2-2E.

  -- A caution note begins: "If the LOCA was ES with loss of MUP. . . ." This is confusing.

  -- The small-break LOCA procedure was the result of misinterpreted instructions issued in May 1978 from B&W concerning a worst case small-break LOCA. The procedure as written is illogical and confusing and cannot be followed.

- Step 3.2 requires that a site emergency be declared for any leak or rupture that results in ESF actuation.

- Following action does not make any provision for shifting to low pressure injection or shifting injection pump suction to the reactor building sump when the borated water storage tank goes dry.

- The procedure requires that engineered safeguards be bypassed regardless of the seriousness of the leak or rupture in order to prevent exceeding HPI pump runout. This means, in effect, that avoiding runout limits is more important than continuing high pressure injection at the maximum rate.

- The procedure includes no cautionary note or other guidance concerning ensured adequate core cooling.

- The procedure is silent with regard to when HPI can be secured if it initiated automatically.

In summary, the LOCA/loss-of-pressure emergency procedure may not be adequate to ensure that the integrity of the core will be maintained in case of a LOCA.

Appendix A is a copy of operating procedure 2202-1.5 and

Appendix B is a copy of emergency procedure 2202-1.3 for reference.

Appendix C is a copy of a portion of the loss-of-reactor coolant procedure from another utility. Included for comparison is a list of cases considered by the procedure and a copy of Case A7: "Small Break -- No Feedwater-No RC Pumps-Reactor Trip," a worst-case small-break situation.

FINDINGS

Analysis of the technical aspects of the operating, abnormal, and emergency procedures that were used or that were applicable on March 28, 1979, at TMI-2 suggests the following findings:

- Although they may be deficient in minor respects, the procedures listed below are adequate for their intended purpose:

  - Operating procedure 2102-2.1, power operations
  - Operating procedure 2104-1.3, decay heat removal system
  - Operating procedure 2102-3.3, decay heat removal via OTSG
  - Operating procedure 2104-1.1, core flooding system
  - Operating procedure 2104-1.4, reactor building spray
  - Operating procedure 2104-6.3, emergency feedwater
  - Operating procedure 2105-1.3, safety features actuation system

- The following procedures contain significant deficiencies which could cause confusion or lack of action but would not preclude their use by thinking operators:

  Operating procedure 2103-1.4, reactor coolant pump operation.

  Precludes pump operation with excessive vibration.

  Whether pump should be tripped under low pressure, LOCA condition was not clear.

  Abnormal procedure 2203-2.2, turbine trip.

  Does not require operator to verify that the PORV is shut although it is expected to open.

  The operator is directed to use let down, as necessary, to preclude pressurizer level from exceeding 240 inches following a turbine trip.

  Emergency procedure 2202-2.2, loss of steam generator feed:

  Requires immediate manual reactor trip on loss of both feedwater pumps.

  Does not require verification of proper PORV operation.

Emergency procedure 2202-1.1, reactor trip:

Makes no provision for determining the cause of the reactor trip and correcting it.

° The following procedures were so deficient as to be inadequate:

Operating procedure 2103-1.3, pressurizer operation:

States the pressurizer may not be taken solid for any reason except hydrostatic tests.

-- Abnormal procedure 2203-2.6, post-accident hydrogen control:

Does not recognize the rapid generation of hydrogen as occurred at TMI.

Does not recognize any difficulties which might be encountered in placing the hydrogen recombiner in operation.

Emergency procedure 2202-1.5, pressurizer system failures:

- Basic structure is very confusing; some sections should be in the loss-of-coolant procedure; symptoms are significantly incomplete, misleading, or erroneous.

No guidance is given for actual pressurizer level control problems.

Terminology is sloppy.

-- Emergency procedure 2202-1.3, loss-of-reactor-coolant/reactor coolant system pressure:

Procedure lacks objectives.

Symptoms are incomplete, misleading, or erroneous.

Procedure is difficult to use. Cases are not defined.

The operator is required to throttle HPI to prevent pump runout, regardless of the severity of the accident.

Procedure does not promptly ensure that containment is isolated.

A section on small-break LOCA response is illogical and cannot be followed.

No cautionary guidance is included regarding core covering and cooling.

- Operators were prohibited by the technical specifications from permitting the pressurizer to go solid.

- Some procedures emphasize avoiding equipment damage over keeping the core covered with water or maintaining core cooling.

- The procedure for decay heat removal via the OTSG is simple, straightforward and, if followed, can be used to cool the core either with or without running reactor coolant pumps.

- Procedures recognize that the PORV will open following a turbine trip.

- Procedures in general are written to minimize "outage" time and maximize "plant availability."

- The turbine trip procedure requires that if the cause of the trip was loss of feedwater flow the operator should verify emergency feedwater flow to the steam generators.

- In addition to the major, substantive deficiencies cited in the analysis, Metropolitan Edison procedures contain many minor errors in substance, typographical errors, imprecision or sloppiness in terminology, and format deficiencies, and the like, which reflect a lack of quality essential in nuclear power plant procedures.

- In reviewing Three Mile Island procedures, there is no evidence that operating experience (lessons learned) at the Island or at other Babcock & Wilcox plants was incorporated into operating procedures.

ACRONYMS

ABP	abnormal procedure
BWST	borated water storage tank
CFR	Code of Federal Regulations
DHRS	decay heat removal system
EF	emergency feedwater
EP	emergency procedure
ES/ESF	engineered safety features
FA	fuel assembly
FSAR	Final Safety Analysis Report
FW	feedwater
HPI	high pressure injection
LOCA	loss-of-coolant accident
MS	main steam
MU	make-up
MUP	make-up pump
NRC	Nuclear Regulatory Commission
OP	operating procedure
OTSG	once-through steam generator
PORV	pilot-operated relief valve
RB	reactor building
RC	reactor coolant
RCS	reactor coolant system
SFAS	safety features actuation system

APPENDIX A

CONTROLLED COPY
TRAINING DEPT.

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.5
PRESSURIZER SYSTEM FAILURE

Table of Effective Pages

Page	Date	Revision	Page	Date	Revision	Page	Date	Revision
1.0	06/22/77	1	26.0			51.0		
2.0	06/22/77	1	27.0			52.0		
3.0	06/22/77	1	28.0			53.0		
4.0	11/17/77	2	29.0			54.0		
5.0	09/29/78	3	30.0			55.0		
6.0	03/11/77	0	31.0			56.0		
7.0	03/11/77	0	32.0			57.0		
8.0	03/11/77	0	33.0			58.0		
9.0	03/11/77	0	34.0			59.0		
10.0	03/11/77	0	35.0			60.0		
11.0	03/11/77	0	36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
20.0			45.0			70.0		
21.0			46.0			71.0		
22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval Approval NA Date — Cognizant Dept. Head	Unit 2 Staff Recommends Approval Approval NA Date — Cognizant Dept. Head
Unit 1 PORC Recommends Approval NA Date — Chairman of PORC	Unit 2 PORC Recommends Approval J F Ihlenfeld Date 9/28/78 Chairman of PORC
Unit 1 Superintendent Approval NA Date —	Unit 2 Superintendent Approval G K Miller Date 9/27/78

Manager Generation Quality Assurance Approval NA Date —

THE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.5
PRESSURIZER SYSTEM FAILURE

SECTION A Leaking Pilot Operated (electromatic) Relief Valve (RC-R2)

A.1 SYMPTOMS

1. Relief valve discharge line temperature exceeding the normal 130°F. Alarms on computer at 200°F.
2. RC drain tank pressure above normal on the control room radwaste disposal control panel and temperature above normal on the local radwaste disposal control panel.
3. RC System makeup flow above normal for the variable letdown flow and RC pump seal in-leakage conditions.
4. Boric Acid concentration continually increasing in the pressurizer.

A.2 IMMEDIATE ACTIONS

A. Automatic Actions

1. None.

B. Manual Actions

1. Close the Electromatic Relief Isolation Valve, RC-V2.

A.3 FOLLOW-UP ACTION

1. Repair during next shutdown.
2. Limit rate of change on ICS to less than 1% per minute while RCV is closed except for runbacks.

SECTION B Inoperative Pilot operated (electromatic) Relief Valve (RC-R2)

B.1 SYMPTOMS

1. RC System pressure is above 2255 psig and RC-R2 fails to open.
2. RC System pressure is below 2205 psig and RC-R2 fails to close.
3. RC-R2 discharge line temperature is above the 200 °F alarm. Computer Point (402)
4. The RC drain tank pressure and temperature are above normal on the control room radwaste disposal control panel 8A.

B.2 IMMEDIATE ACTION

A. Automatic Action

1. For a failed closed RC-R2:
   a. Pressurizer heaters off at 2160 psig. Spray valve RC-VI is open above 2205 psig.
   b. Reactor trip occurs at 2355 psig.
   c. Pressurizer code relief valves open at 2450 psig.
2. For a failed open RC-R2:
   a. All pressurizer heater banks on full below 2105 psig.
   b. Reactor trips at 1900 psig or variable pressure temperature.
   c. High Pressure Injection is actuated at 1600 psig.

B. Manual Action

1. For a failed close RC-R2:
   a. Shift spray valve RC-VI to "MANUAL" and open further for additional spray flow.
   b. Insure all pressurizer heaters off above 2160 psig.
   c. If reactor power is being changed (except for a runback) stop the power change until pressure is returned to normal.
   d. Isolate RC-R2 by closing RC-V2.
2. For a failed open RC-R2:
   a. Close Electromatic Relief Isolation Valve (RC-V2).
   b. Insure all pressurizer heaters on below 2105 psig.

B.3 FOLLOW-UP ACTION

1. Return system pressure and temperature to normal.
2. Reduce ICS Rate of Change to less than 1% per minute (except for Runbacks.)

SECTION C Leaking Code Relief Valve (RC-R1A or RC-R1B)

C.1 SYMPTOMS

1. Code relief valve discharge line temperature(s) exceeding the computer normal 130°F. Computer alarms at 200°F. Computer Point (403) (404)
2. RC drain tank pressure and temperature above normal on the control room radwaste disposal control panel 8A.
3. RC System makeup flow is above normal for the variable letdown flow and RC pump seal in-leakage conditions.
4. Boric Acid Concentration continually increasing in the Pressurizer.

C.2 IMMEDIATE ACTION

A. Automatic Action

1. None.

B. Manual Action

1. Determine RC leakage according to 2301-3D3.

C.3 FOLLOW-UP ACTION

1. If RC system identified leakage is in excess of 10 gpm, reduce the leakage rate to within limits within 4 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.
2. It will be necessary to recirculate the pressurizer through the spray valve to equalize Boron concentration.
3. Place Code Relief Discharge Line temperatures on Analog Trend Recorder.

SECTION D Inoperative Code Relief Valve (RC-R1A or RC-RIB)

D.1 SYMPTOMS

1. Code relief valve(s) fail to open when RC pressure is above 2450 psig.
2. Code relief valve(s) fail to close when RC pressure is below 2325 psig.
3. Code relief valve(s) discharge line temperature is above 200 °F alarm.
4. The RC Drain Tank pressure and temperature are above normal on the control room radwaste disposal control panel.
5. RC system makeup flow is above normal for the variable letdown flow and the RC pump seal in-leakage conditions.

D.2 IMMEDIATE ACTION

A. Automatic Action

1. For a fail to open code relief valve;
   a. Reactor trip occurred at 2355 psig.
   b. Spray valve RC-V1 opened above 2205 psig.
2. For a fail to close code relief valve:
   a. Reactor trip occurs at 1900 psig or on variable P/T.
   b. Increased makeup flow.
   c. All pressurizer heaters energized.
   d. Safety Injection is actuated at 1600 psig.

B. Manual Action

1. For a fail to open code relief valve;
   a. Place pressurizer spray valve in "MANUAL" and open further for additional spray flow.

b. Verify pressurizer heaters are "OFF" at plant control panel.

2. For a fail to close code relief valve:

a. Turn all heaters "ON" at plant control panel.

b. Isolate letdown flow at plant control panel by "CLOSING" MU-V376.

c. Open DH-V5A. Start MU-P1A if necessary. Attempt to control pressurizer level using MU-V16B.

d. Manually initiate safety injection if required to maintain pressurizer level.

D.3 <u>FOLLOW-UP ACTION</u>

1. For a fail to open code relief valve:

a. Proceed with cooldown.

2. For a fail to close code relief valve:

a. Hold pressurizer, if possible, at or greater than 220 inches with Safety Injection.

b. Proceed with cooldown.

c. With no pressurizer code safety valve operable, immediately suspend all operations involving positive reactivity changes and place an operable DHR Loop into operation in the shutdown cooling mode.

d. With a pressurizer code safety valve inoperable, either:

1. Restore the inoperable valve to operable status within 15 minutes or

2. Be in Hot Shutdown within 12 hours.

SECTION E Inoperative Pressurizer Heaters

E.1 SYMPTOMS

1. Heater banks fail to energize or de-energize if RC pressure is at heater bank setpoint.

	Bank 1	Bank 2	Bank 3	Bank 4	Bank 5	Units
ON	2150	2145	2135	2120	2105	PSIG
OFF	2160	2155	2155	2140	2125	PSIG

NOTE: Banks 1, 2, and 3 are full on at "ON" setpoint.

2. Pressurizer level Lo-Lo alarm at 80 inches.
3. Pressurizer heater power supply ground alarm.
4. Abnormal console indicating lights for the heating groups.
5. High (2255 psig) or low (2055 psig) pressure alarms.

E.2 IMMEDIATE ACTION

A. Automatic Action

1. For energized heaters and rising presssure:
   a. Pressurizer spray valve (RC-V1) open (red and green console jog button lights).
2. For loss of heaters and decreasing pressure: None.

B. Manual Action

1. If control malfunction is suspected:
   a. Place heater controller in "MANUAL".
2. For energized heaters and rising pressure:
   a. Attempt to de-energize all heaters excepts Banks 1 or 2. (Groups 12 or 13 respectively)
3. For loss of heaters and decreasing pressures:
   a. Attempt to energize backup heaters from plant control panel.
   b. If a. is unsuccessful, being reducing unit load.

E.3 FOLLOW-UP ACTION

1. For energized heaters and rising pressure:
   a. Open heater breakers in question at the pressurizer heater control centers except for Banks 1 or 2 (Groups 12 or 13 respectively).
   b. Control RC pressure at the normal 2155 psig set point with the pressurizer spray valve (RC-V1) in "MANUAL."
2. For loss of heaters and decreasing pressure:
   a. Determine cause.
   b. If pressure cannot be maintained with the remaining heaters, continue load reduction to shutdown and possibly cooldown condition.
   c. Close RC-V3 and reopen periodically to maintain spray line temperature greater than 540°F.

SECTION F Malfunction In Pressurizer Level Indication or Control

F.1 SYMPTOMS

1. Disagreement between the console recorder level readouts of more than 12 inches.
2. Rapid change in indicated/recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction.

F.2 IMMEDIATE ACTION

A. Automatic Action

1. If indication fails low:
   Pressurizer heaters trip at 80 inches, makeup value MU-V17 opens, and RC pressure increases.
2. If indication fails high;
   Makeup valve MU-V17 closes.

B. Manual Action

1. When any two of three console recorder level transmitter readouts disagree by more than 12 inches, take manual control of level and then select the third transmitter for indication.
2. Re-energize heaters if tripped due to malfunction.

F.3 FOLLOW-UP ACTION

1. If the switching level transmitters has not rectified the condition, switch to the alternate temperature detector.
2. If pressurizer level recorder indication is lost, select another transmitter or use the computer for level indication.

SECTION G Pressurizer Spray Valve Failure (RC-V1)

G.1 SYMPTOMS

1. Pressurizer spray valve (RC-V1) fails to open when the RC system pressure is greater than 2205 psig.
2. Pressurizer spray valve (RC-V1) is open when the RC System is less than 2155 psig.

G.2 IMMEDIATE ACTION

A. Automatic Action

1. RC system pressure greater than 2255 psig activates RC-R2 electromatic relief and the high pressure alarm.
2. RC-V1 failing open (in auto) causes RC system pressure to stablize at approximately 2100 psig with all heaters "on".
3. Failure when manually opened beyond the automatic limit position causes continued pressure drop and alarm at 2055.

B. Manual Action

1. Control RC-V1 opening or closing in "MANUAL" with jog buttons.
2. If the spray valve has failed open, control pressure by closing the pressurizer spray isolation valve (RC-V3).

NOTE: If the pressurizer spray isolation valve (RC-V3) is closed, it must be periodically cycled to keep the spray line warm. Cycle RC-V3 is open as necessary to stay above RC pressurizer spray line temperature alarm of 540 °F. (Computer point 0405).

CAUTION: Do not exceed at AT of 410°F between pressurizer temperature and reactor coolant hot leg temperature.

3. Reduce rate of ICS load change to less than 1% per minute.

G.3 FOLLOW-UP ACTION

1. Continue plant operation with reduced rate of load change.
2. Check thermal overload on RC-Vl and reset if necessary.

APPENDIX B

2202-1.3
Revision 11
10/06/78

CONTROLLED COPY
CENTRAL FILE

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.3

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM PRESSURE

Table of Effective Pages

Page	Date	Revision	Page	Date	Revision	Page	Date	Revision
1.0	06/22/77	1						
2.0	05/12/78	8						
3.0	12/30/77	3						
4.0	12/30/77	3						
5.0	06/22/77	1						
6.0	03/01/78	5						
7.0	09/29/78	10						
8.0	05/12/78	8						
9.0	06/20/78	9						
10.0	05/12/78	8						
10.1	05/12/78	8						
10.2	05/12/78	8						
11.0	06/22/77	1						
12.0	06/22/77	1						
13.0	04/18/78	7						
14.0	10/06/78	11						
15.0	04/18/78	7						
1[illegible].0	12/30/77	3						
[illegible]	06/22/77	1						
[illegible].0	04/18/78	7						
19.0	12/30/77	3						
20.0	12/30/77	3						
21.0	12/30/77	3						
22.0	12/30/77	3						
23.0	06/22/77	1						

Unit 1	Unit 2
Unit 1 Staff Recommends Approval Approval NA Date —— Cognizant Dept. Head	Unit 2 Staff Recommends Approval Approval NA Date —— Cognizant Dept. Head
Unit 1 PORC Recommends Approval NA Date —— Chairman of PORC	Unit 2 PORC Recommends Approval RP Warren Date 10/6/78 V - Chairman of PORC
Unit 1 Superintendent Approval NA Date ——	Unit 2 Superintendent Approval J L Seelinger Date 10/6/78

Manager Generation Quality Assurance Approval NA Date ——

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.3

PRESSURE

A. <u>Leak or Rupture Within Capability of System Operation.</u>

1.0 <u>SYMPTOMS</u>

1.1 Initial loss of reactor coolant pressure & decrease in pressurizer level becoming stable after short period of time.

1.2 Possible reactor building high radiation and/or temp. alarm.

1.3 Possible reactor building sump high level alarm.

1.4 Mike-up tank level decreasing >1" in 3 min.

1.5 Possible make-up line high flow alarm.

1.6 RB Fan Drip Pan Level Hi Alarms.

NOTE: The operator may distinguish between a loss of coolant inside containment, an OTSG tube rupture and a steam line break by the following symptoms which are unique to the aforementioned accidents.

1. Loss of coolant inside Rx Bldg. - particulate, iodine & gas monitor alarm on HP-R-227 "Reactor Building Air Sample."
2. OTSG tube rupture - gas monitor alarm on VA-R-748.
3. Steam line break
   (1) Low condensate storage tank level alarm - and or low hot well level alarm.
   (2) FW Latch System Actuation.

2.0 <u>IMMEDIATE ACTION</u>

2.1 Automatic Action:

2.1.1 KU-V17 will open to compensate for reduced pressurizer level.

2.1.2 Additional pressurizer heaters will come on in response to reduced reactor coolant pressure.

2.2 Manual Action

2.2.1 Verify MU-V17 open and pressurizer heaters on.

2.2.2 "CLOSE" MU-V376 let-down isolation valve, & "START" the backup MU pump, if required.

2.2.3 Reduce load at 10 percent minute & proceed with normal shutdown.

2.2.4 "LINE-UP" waste transfer pump from a R.C. Bleed Holdup Tank & pump to the make-up tank to maintain required level.

2.2.5 <u>If for any reason the operator cannot maintain Make-up Tank- and Pressurizer levels above their respective low level alarm setpoints, "TRIP" the reactor, "INITIATE" Safety Injection manually (push buttons on panel 3), & then</u> "Close" MU-V12

3.0 <u>FOLLOW UP ACTION</u>

3.1 Safety Injection Not Initiated.

3.1.1 Initiate unit shutdown & cooldown per 2102-3.1 and 2102-3.2 respectively.

3.2 Safety Injection Manually Initiated (HPI and LPI).

3.2.1 Verify that the Make-up Pumps & Decay Heat Removal Pumps start satisfactorily.

3.2.1.1 Close MU-V12 and MU-V18.

3.2.2 Bypass the SAFETY INJECTION by DEPRESSING the Group Reset Pushbuttons & "THROTTLE" MU-V16A/B/C/D as necessary to maintain 220" pressurizer level and not exceed 250 GPM/HPI flow leg.

3.2.3 If MU pump flow drops below 95 GPM, trip excess MU pumps.

NOTE: HPI String A flow is the sum of MU23 FE1&2. HIP String B flow is the sum of MU23 FE3&4.

3.2.4 Verify that Safety Injection equipment is in its ESF position as shown in Table A-1.

3.2.5 CAUTION: Continued operation depends upon the capability to maintain pressurizer level and RCS pressure above the 1,640 PSIG Safety Injection Actuation setpoint.

1. If pressurizer level can be maintained above the low level alarm point and the RCS pressure above the Safety Injection Actuation point, then proceed to step 3.2.6.
2. If pressurizer level cannot be maintained above the low level alarm point and the RCS pressure above the Safety Injection Actuation point, then the plant has suffered a major rupture and operation should continue according to Part B - Leak or Rupture of Significant Size Such that Engineered Safety Features Systems are Automatically Initiated.

3.2.6 With the pressurizer level and RCS pressure being maintained within allowable limits, initiate plant shutdown and cooldown per 2102-3.1 and 2102-3.2, respectively.

NOTE: The HPI System is being used for makeup control and valves MU-V16A/B/C/D will have to be throttled to maintain pressurizer level. As RCS pressure decreases, it may be possible to return to the normal make-up flowpath and secure HPI. If MU pump flow drops below 95 GPM as a result of throttling, "Open" MU-V36 & 37 to provide MU pump recirculation path to MU Tank. Monitor MU Tank level and open MU-V12 as required.

3.2.7 At the time the DH System is to be brought on line for normal cooling only one DH string should be used for decay heat removal (i.e. - recirculation from the RC System). The other DH string should be maintained on standby for use in recirculating water from the RB sump to the RC system.

<u>NOTE:</u> Trip Reactor Coolant Pumps before RC Pressure decreases below pump NPSH (See figure 1 of 2102-3.1/3.2.

3.2.8 When the Borated Water Storage Tank level decreases to 12' as indicated on panel 8, shift the MU/HPI pump(s) suction from the BWST to the RB sump if RCS Pressure is greater than 200 psig as follows: (assume DH string A(B) is being used for decay heat removal and DH string B(A) is being maintained on standby):

3.2.8.1 If not already done, "THROTTLE" HPI string(s) flow rate to at least 500 gpm each (250 gpm per leg) using control valves MUV16A/B/C/D (or MU-V17 if flow has been returned to the normal make-up flow path). Flow rate indication and valve control in control room on Panel 8 and 3, respectively.

3.2.8.2 "OPEN" valve DH-V7B(A) in crossover line from LPI String B(A) to HPI string B(A) (suction of HPI pumps).
"REPOSITION" HPI flow control valves MU-V16A/B/C/D (or MU-V17 if flow has been returned to the normal make-up flowpath). HPI flow would increase because of increased HPI pump suction pressure.

3.2.8.3 When the BWST level decreases to 7', verify automatic transfer to the RB sump is initiated. Verify OPEN suction valve for string B(A), DH-V6B(A) from the RB sump.

3.2.8.4 When the suction valve from the RB sump DH-V6 B(A), is fully open, then "CLOSE" the ECCS suction valve, DH-V5B(A), from the BSWT (valve controls and position indication in control room). The ECCS B(A) string is now in "piggy-back" operation providing makeup to the RCS from the RB sump as required.

3.2.9 After RCS pressure decreases to = 200 psig, throttle HPI discharge flow by throttling MU-V16A/B/C/D. Observe that LPI pumps now deliver water to RCS via DH-V4A/B.

3.2.10 When MU-V16A/B/C/D (HPI flow valves) are closed, stop the Hi pressure injection pumps & close DH-V7A & 7B from the LPI pump discharge. Injection flow path is now as follows:

Spill coolant to RB sump, RB sump to LPI pumps,
LPI pumps to RCS via DH-V4A/B.

3.2.11 Throttle DH-V128A & B as required to maintain 220" pressurizer level and max. LPI pump flow of 3,000-3,300 gpm. Within about 24 hours, establish a long-term cooling circulation mode as described in 2104-1.3 and listed below.

Mode 1 Forced circulation using decay heat drop line.

Mode 2 Gravity draining reactor coolant hot leg to the Reactor Building sump via the DH drop line.

Mode 3 Hot leg injection using Pressurizer Auxiliary Spray Line.

Mode 4 Reverse flow through the Decay Heat Drop line into "B" Reactor Coolant Loop Hot Leg.

3.2.12 Evaluate radiation levels & initiate action for Site Emergency as outlined in the TMI radiation emergency plan.

3.2.13 Reactor Building Isolation Initiated

1. Refer to Section B, 3.0 & complete all steps.

B. LEAK OR RUPTURE OF SIGNIFICANT SIZE SUCH THAT ENGINEERED SAFETY FEATURES SYSTEMS ARE AUTOMATICALLY INITIATED.

1.0 SYMPTOMS

1.1 Rapid continuing decrease of reactor coolant pressure.

(1) Lo alarm 2,055 psig.

(2) Lo-Lo-alarm 1700 psig.

(3) Safety Injection actuation at 1,640 psig.

1.2 Rapid continuing decrease of pressurizer level.

(1) Lo alarm 200".

(2) Lo-Lo alarm 80" (Interlock heater shutoff).

1.3 Hi radiation alarm in Reactor Building.

1.4 Reactor Building Ambient Temperature Alarm.

1.5 Hi Reactor Building Sump level.

1.6 Hi Reactor Building pressure (RCS or main steam line rupture).

1.7 Rapidly decreasing make-up tank level.

1.8 Both core flood tanks levels & pressures are decreasing.

NOTE: The operator may distinguish between a loss of coolant inside containment, an OTSG tube rupture and a steam line break by the following symptoms which are unique to the aforementioned accidents.

1. Loss of coolant inside Rx Bldg. - particulate, iodine gas monitor alarm on HP-R-227 "Reactor Building Air Sample."

2. OTSG tube rupture - Gas monitor alarm on VA-R-748.

3. Steam break inside Rx Bldg:

(1) Low condensate storage tank level alarm - and or low hot well level alarm.

(2) FW Latch System Actuation.

2.0 IMMEDIATE AC.ION

2.1 Automatic Action.

2.1.1 Reactor trip 1,900 psig.

2.1.2 Turbine Trip.

2.1.3 Safety Injection initiated @ 1,600 psig RCS pressure, or 4 psig Reactor Building pressure.

2.1.4 Both Core Flood Tank levels & pressures may decrease depending upon rupture size and RCS pressure. (<600 psig).

2.1.5 Reactor Building Isolation & Cooling initiated. (R.B. Press. >4 psig).

2.1.6 Reactor Building Spray if the Reactor Building pressure is greater than 30 psig.

2.2 Manual Action.

2.2.1 "CLOSE" KU-V12 and MU-Vl8.

2.2.2 Small-break LOCA Response.

2.2.2.1 Within 2 minutes of the LOCA the CRO dedicated to recognition of a small break LOCA must complete the following:

a. Verify that small-break LOCA with single failure symptoms exist.

Symptoms: 1. SFAS initiation and only one MUP started, or

2. SFAS initiation and loss of 2-lE or 2-2E

b. DISPATCH designated LOCA Response Primate A.0. to OPEN MUP Discharge Cross-connect.

c. PROCEED to MU-V16A & B or MU-V16C & D.

d. Within 5 minutes of the LOCA the MUP discharge cross connect valve must be opened off its closed seat and one of the MU-V16 valves on the side of the single failure must be opened 2 turns.

2.2.2.2 CRO at Ml'-V16A & B or MU-V16C & D must ESTABLISH communications with the Control Room.

2.2.2.3 Once in communication with the control room the CRO at MU-V16A & B or C & D continue to open the valves to establish 125 gpm per leg; while the control room CRO THROTTLES MU-V16C & D or A & B to prevent pump runout.

NOTE: If the LOCA was E.S. with loss of MUP all MU-V16 valves would initially be open. The operator at the controls must in this case throttle all 4 MU-V16 valves to 125 gpm/leg. Also in this case the CRO dedicated to small break LOCA would not need to go to the MU-V16 valves. He should go to the MUP discharge cross connect valves and assist the A.O. to speed up the opening of the MUP discharge cross connect.

SMALL BREAK LOCA ACTION TIMES

EVENT	TIME from Occurance
Recognition	< 2 minutes CRO to
MU-16's	- 4.5 minutes

SMALL BREAK LOCA ACTION TIMES (Cont'd) 06/20/79

EVENT	TIME From Occurance
AO to MUP Discharge X-Connects	< 3.5 minutes
Communications Established With CRO AT MU-V16's	< 5.0 minutes
One of the Single Failure Side MU-V16's Open 2 turns	< 5.0 minutes
Discharge X-Connect OFF Closed Seat	• 5.0 minutes
Discharge X-Connect Open	• 10.0 minutes
MU-Vl6's Throttled to 125 gpm/leg	• 10.0 minutes

2.2.3 Verify Hi pressure injection is operating properly as evidenced by injection flow in all four legs (MU-V16A/B/C/D). Flow indicated on MU23 FE1,2,3,4.

2.2.4 "TRIP" reactor coolant pumps before reaching 1,200 psig.

2.2.5 Verify Reactor Building Cooling and Isolation is operating properly.

3.0 FOLLOW UP ACTION

3.1 Verify that all E.S.F. equipment is in its ESF position, by observing that all equipment status lights indicate as shown in Table B-1.

3.1.1 Check locked valve status book and verify closed or close the following manual containment isolation valves MU-V330, MU-364, CF-V114A, CFV114B, CF-V145, CF-V146, DH-V187, and DW-V28.

NOTE: Should any component not operate properly, attempt to actuate it at its remote switch in the Control Room. If it still does not operate, & the component has a local control station attempt to operate the component locally.

3.2 Notify Shift Foreman, who notifies all Station personnel over the cross-tied PA system that a site emergency has occurred.

3.3 0 to 20 or 30 minutes past LOCA until sump recirculation is initiated: Control Room operator continuously monitors the following:

9.0

3.3.1 Liquid levels in the:

1. Borated Water Storage Tank, DH-T1, (DH-3-LI 1/2).
2. Sodium Hydroxide Tank, DH-T2, (DH-7-LI).

3.3.2 Safety Features flow rate in each of the following:

1. Two Low Pressure (Decay Heat) Injection lines, DH-l-FI 1 and 2.
2. Four High Pressure (Make-up) Injection lines, MU-23-FI-1, 2, 3, and 4.
3. Two Reactor Building Spray injection lines, BS-1-FI and 2.
4. Four of five reactor building emergency cooling river water lines AH-FI-5620, 5621, 5522, 5623, or 5624 respectively.

3.3.3 Reactor Building environmental indications:

1. Temperature, recorder on Panel 25.
2. Pressure, recorder on Panel 3.

3.4 "DEFEAT" any two channels of Reactor Building Isolation and Cooling, then bypass all three Safety Injection Channels.

CAUTION: If normal power is lost while operating in the injection Mode from the BWST, RB Isolation and Cooling must be manually initiated, when either the BUS 2-lE or 2-2E Undervoltage alarm is received to ensure proper diesel generator load sequencing.

3.5 "THROTTLE" as required to prevent pump runout:

1. High press. inj. flow (MU-16A/B/C/D) 0-250 GPM/LEG.

   CAUTION: If MU pump flow drops below 95 GPM trip the excess MU pumps.

2. Low Press. Inj. Flow (DH-V128A/B) 3000-3300 GPM/PMP.
3. Building spray flow (BS-V1A/B) 1400-1700 GPM/PMP.

<u>NOTE:</u> Hi flow alarms should actuate as a warning to throttle flows.

<u>CAUTION:</u> The actions to be taken for switching suction from the BWST to the R.B. sump depend upon the number of operating ECCS injection strings and the delivered flowrates in these injection strings. Based upon the existing situation, in the ECCS, proceed as outlined below to perform switch over of suction to the RB sump:

	Situation	Go to Step
1.	Both LPI strings are operating and indicated flow in each is above 750 gpm.	3.6
2.	Both LPI strings are operating but indicated flow in each is below 750 gpm.	3.7
3.	One LPI string is inoperative.	3.8

<u>NOTE:</u> The main objective when switching suction from the BWST to the RB sump is to maintain ECCS flow through two flow paths.

3.6 Both LPI Strings are Operating and Indicated Flow in Each is Above 750 GPM.

3.6.1 When the BWST level reaches approximately 12', initiate the following steps:

3.6.2 If not already done "THROTTLE" LPI strings flow rates back to 3000 GPM each using control valves DH-V128 A & B (flow rate indication and valve control in control room).

3.6.2.1 If not already done, THROTTLE BS pump's flows back to 1,600 gpm per pump. This must be done prior to taking suction from the RB sump.

3.6.3 "SHUT OFF" HPI pumps (pump control in control room).

3.6.4 Verify the ECCS suction valves DH-V6A & B from RB sump automatically open at BWST level of 7'.

3.6.5 When suction valves from RB sump are full open, "CLOSE the ECCS suction valves (DH-V5A & B) from the BWST.

3.6.6 "REPOSITION" LPI flow control valves (DH-V128 A & B) to obtain 3000 GPM each string if necessary. (Flow rate could change due to change in suction sources.)

3.6.7 Proceed to step 3.9.

3.7 Both LPI Strings Are Operating But Indicated Flow In Each Is Below 750 GPM.

3.7.1 When the BWST level reaches approximately 12', initiate the following steps.

3.7.2 If not already done, "THROTTLE" HPI strings' flow rates back to 500 GPM per pump each using control valves MU-V16A, B, C, and D (flow rate indication and valve control in control room).

3.7.2.1 If not already done, THROTTLE BS pump's flows back to 1,600 GPM per pump. This must be done prior to taking suction from the RB sump.

3.7.3 "OPEN" valves DH-V7A and B crossover line from LPI line to suction of HPI pumps (valve control and position indication in control room). "Reposition" HPI flow control valves (MU-V16A, B, C, & D) to obtain 500 GPM each string. (HPI flow would increase because of increased HPI pump suction pressure).

3.7.4 VERIFY the ECCS suction valves (DH-V6A & B) from RB sump automatically open at BWST level of 7'.

3.7.5 When suction valves from RB sump are fully open, "Close" the ECCS suction valves (DH-V5A & B) from the BWST. The ECCS is now in "piggy-back" operation.

3.7.6 Proceed to step 3.9.

<u>NOTE:</u> Once the flow in each LPI string exceeds 750 gpm, the HPI pumps can be "SHUT OFF" and valves DH-V7A & B can be "CLOSED."

3.8 One LPI String is Inoperative

3.8.1 The BWST 7' automatic transfer to the RB sump is reached in approximately 55-80 minutes from initiation of ECCS injection, depending upon reason for string failure (i.e. - local LPI failure or diesel failure). Prior to actuation of the to-lo level alarm, initiate the following steps.

3.8.2 Using the controls in the control room, attempt to "START UP" the non-operating LPI String. If successful, proceed to step 3.6. If not successful, proceed to step 3.8.3 below.

3.8.3 If step 2 was unsuccessful, initiate opening the DH cross-connect isolation valves (DH-Vl93 A & B) as follows:

3.8.3.1 ENSURE SN-V188 is closed then OPEN DH-V112 A & B to fill the inoperable string.

NOTE: If off-site power is lost DHV193 A & B must be manually opened.

3.8.3.2 Obtain the keys for the DH cross-connect isolation valves' (DH-V193A and DH-V193B) breakers from the shift supervisor.

3.8.3.3 Proceed to 480V MCC 2-32B and MCC 2-42B.

3.8.3.4 Remove the locks from the isolation valve breakers for the DH cross-connect line.

NOTE: Local control stations for DH-V193A and are located in the Aux Bldg at Elev. 280"6" near the DH vaults.

3.8.3.5 "OPEN" the DH cross-connect isolation valve (e.g. DH-V-193A(B)) next to the operating LPI String.

3.8.3.6 "OPEN" the second isolation valve (DH-V193A(B)).

3.8.3.7 While opening the second decay heat cross-connect isolation valve, "THROTTLE" either DH-V128A or DH-V128B in the control room as required to achieve essentially equal flow rates in both DH injection lines. (Approximately 1,500 gpm per LPI string).

3.8.3.8 If flow is established at greater than 750 gpm through each LPI string, then proceed to step 3.8.5.

3.8.3.9 If flow cannot be established through each LPI string in excess of 750 gpm using the cross-connect line before ECCS suction must be switched to the RB sump, then proceed to step 3.9.4.

3.8.4 If opening the DH cross-connect line fails to provide flow in each LPI string in excess of 750 gpm, then place one HPI string in a modified "piggy-back" mode with the operating LPI string as follows (assume LPI string "A"(B) is the operating stiring):

3.8.4.1 If not already done, "THROTTLE" HPI string "A"(B) flow rate back to 500 GPM using control valve MU-V16A and V16B.

3.8.4.2 "THROTTLE" LPI string "A"(B) flow rate to 3,000 GPM using control valve DH-V128A(B).

3.8.4.3 "OPEN" valve DH-V7A(B) in crossover line from LPI line to suction of the operating HPI pump.

3.8.4.4 "REPOSITION" HPI flow control valves MU-V16A&B to obtain 250 GPM per leg HPI flow and reposition LPI flow control valve DH-V128A to obtain 2,500 GPM LPI flow. The LPI pump is pumping design flow of 3,000 GPM (2,500 GPM LPI plus 500 GPM HPI).

3.8.4.5 "SHUT OFF" HPI pump in HPI string "B"(A).

3.8.4.6 Proceed to step 3.8.6.

3.8.5 "SHUT OFF" HPI pumps (pump control in control room).

3.8.6 If not already done, THROTTLE BS pump's flows back to 1,600 GPM per pump. This must be done prior to taking suction from the RB sump.

3.8.6.1 When the BWST level decreased to 7' VERIFY the ECCS suction valves (DH-V6A & B) from RB sump automatically open.

3.8.7 When suction valves from RB sump are full open (position indication in control room), "CLOSE" the ECCS suction valves (DH-V5A & B) from the BWST.

3.8.8 "REPOSITION" LPI flow control valve(s) (DH-V128A and/or B) as required to obtain proper string flowrates. (Flow rates could change due to change in suction sources).

3.9 When the Sodium Hydroxide Tank level reaches approximately 3 ft. "CLOSE" DH-V8A & B.

3.10 Actuate Environmental Barrier System by opening EB-Vll.

3.11 Within 24 hours of ECCS initiation, establish one of the long-term cooling circulation modes described in 2104-1.3 - Decay Heat Removal System, and listed below:

Mode 1 Forced circulation using decay heat drop line.

Mode 2 Gravity draining reactor coolant hot leg to the Reactor Building sump via the D.H. drop line.

Mode 3 Hot leg injection using pressurizer auxiliary spray line.

Mode 4 Reverse flow through the decay heat drop line into "B" Reactor Coolant Loop Hot Leg.

4.0 LONG TERM ACTION

4.1 Verify all previous actions and carry out additional actions as outlined below.

4.2 Evaluate symptoms and determine if possible the cause of the loss of coolant.

4.3 Secure turbine, feed water, and steam systems when time permits.

4.4 Monitor for H_2 buildup and assure actuation of H_2 recombiner per 2104-6.4, Hydrogen Recombiner Operations.

4.5 Monitor R.B. Sump for pH and add Sodium Hydroxide as required through the decay heat removal system.

4.6 As conditions permit, evaluate unit conditions, and return all nonessential equipment to its normal line up.

NOTE: Refer to the following instructions and procedures for additional information as required.

1. Radiation emergency plan site emergency in the emergency plan.
2. 2104-5.4 - Control Building HVAC.

TABLE A-1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM

PANEL 13

The white light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1A	ON
G2-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON
NR-P-1A (Note 3)	ON
NS-P-1A (Note 4)	ON
TIE-2E2	OPEN
T3E-4E2	OPEN
T11E-21E2	OPEN
DC-V96A	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9A	CLOSED
DH-V4A	OPEN
DH-V5A	OPEN
DH-V8A	OPEN
DH-V100A	CLOSED
DH-V102A	OPEN
MU-P-1B (Note 2)	ON
NR-V40A	OPEN
NS-P-1C	ON
T12-22E-2	OPEN
MU-V28	CLOSED

ACTUATION B

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1B	ON
G22-12 (Note 1)	CLOSED
MU-P-1B (Note 2)	ON
NR-P-1C (Note 3)	ON
NS-P-1B (Note 4)	ON
T2E-1E2	OPEN
T4E-3E2	OPEN
T21E-11E2	OPEN
DC-V96B	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9B	CLOSED
DH-V4B	OPEN
DH-V5B	OPEN
DH-V8B	OPEN
DH-V100B	CLOSED
DH-V102B	OPEN
MU-P-1C	ON
NR-V40B	OPEN
NS-P-1C (Note 4)	ON
T22E-12E-2	OPEN
HY-V55	CLOSED

TABLE A-1

ACUATION A

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V36	CLOSED
MU-V16A	OPEN
MU-V16B	OPEN
DH-P-1A	ON
NR-P-1B (Note 3)	ON
T31E-41E-2	OPEN
NS-V84B	CLOSED
NR-V42A	OPEN

ACTUATION B

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V37	CLOSED
NS-V32	CLOSED
NS-V67	CLOSED
DH-P-1B	ON
MU-V16C	OPEN
MU-V16D	OPEN
NR-P-1D (Note 3)	ON
T41E-31E-2	OPEN
NS-V84A	CLOSED
NR-V42B	OPEN

PANEL 8

Equipment	ESF Position	Status Light Indicator
NS-V83A	OPEN	R
NS-V83B	OPEN	R
NS-V215	CLOSED	G
NS-V216	CLOSED	G

NOTE 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (white).

NOTE 2: MU-P-1B will be running if normal power is available for the Actuation, for the pump that it is selected to backup.
If normal power is lost, MU-P-1B will be running, if the pump that it is selected to backup fails to start or is inoperable.

TABLE A-1

NOTE 3: The NR pump in each header selected for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

TABLE B-1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM

PANEL 13

The white light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION A		ACTUATION B	
SAFETY INJECTION GP. 1		SAFETY INJECTION GP. 1	
Equipment	ESF Position	Equipment	ESF Position
DC-P-1A	ON	DC-P-1B	ON
G2-12 (Note 1)	CLOSED	G22-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON	MU-P-1B (Note 2)	ON
NR-P-1A (Note 3)	ON	NR-P-1C (Note 3)	ON
NS-P-1A (Note 4)	ON	NS-P-1B (Note 4)	ON
TIE-2E2	OPEN	T2E-1E2	OPEN
T3E-4E2	OPEN	T4E-3E2	OPEN
T11E-21E2	OPEN	T21E-11E2	OPEN
DC-V96A	CLOSED	DC-V96B	CLOSED

SAFETY INJECTION GP. 2		SAFETY INJECTION GP. 2	
Equipment	ESF Position	Equipment	ESF Position
NR-V9A	CLOSED	NR-V9B	CLOSED
DH-V4A	OPEN	DH-V4B	OPEN
DH-V5A	OPEN	DH-V5B	OPEN
DH-V8A	OPEN	DH-V8B	OPEN
DH-V100A	CLOSED	DH-V100B	CLOSED
DH-V102A	OPEN	DH-V102B	OPEN
MU-P-1B (Note 2)	ON	MU-P-1C	ON
NR-V40A	OPEN	NR-V40B	OPEN
NS-P-1C	ON	NS-P-1C (Note 4)	ON
T12-22E-2	OPEN	T22E-12E-2	OPEN
MU-V28	CLOSED	HY-V55	CLOSED

TABLE B-1

ACTUATION A

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V36	CLOSED
MU-V16A	OPEN
MU-V16B	OPEN
DH-P-1A	ON
NR-P-1B (Note 3)	ON
T31E-41E-2	OPEN
NS-V84B	CLOSED
NR-V42A	OPEN

R.B. ISOLATION AND COOLING GP. 1	
Equipment	Position
AH-V81	CLOSED
AH-V101	CLOSED
AH-V102	CLOSED
AH-V105	CLOSED
AH-V107	CLOSED
CF-V144	CLOSED
DH-V3	CLOSED
NM-V52	CLOSED
NR-V51A	CLOSED
RR-V2A	OPEN
RR-V2B	
RR-V5A	OPEN
RR-V5B	OPEN

ACTUATION B

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V37	CLOSED
NS-V32	CLOSED
NS-V67	CLOSED
DH-P-1B	ON
MU-V16C	OPEN
MU-V16D	OPEN
NR-P-1C (Note 3)	ON
T41E-31E-2	OPEN
NS-V84A	CLOSED
NR-V42B	OPEN

R.B. ISOLATION AND COOLING GP. 1	
Equipment	Position
AH-E-11D	ON
AH-V80	CLOSED
AH-V103	CLOSED
AH-V104	CLOSED
AH-V106	CLOSED
AH-V108	CLOSED
CF-V115	CLOSED
DH-V2	CLOSED
NM-V104	CLOSED
NR-V51B	CLOSED
RR-V2C	OPEN
RR-V2D	OPEN

TABLE B-1

ACTUATION A		ACTUATION B	
R.B. ISOLATION & COOLING GP. 1		**R.B. ISOLATION & COOLING GP. 1**	
Equipment	Position	Equipment	Position
RR-V5C	OPEN	RR-V6C	OPEN
SV-V55	CLOSED	RR-V6D	OPEN
WDL-V1095	CLOSED	RR-V6E	OPEN
DC-V114	CLOSED	SV-V54	CLOSED
		WDL-V1092	CLOSED
		DC-V103	CLOSED
		DC-V115	CLOSED

R.B. ISOLATION & COOLING GP. 2		**R.B. ISOLATION & COOLING GP. 2**	
Equipment	Position	Equipment	Position
AH-E-4A	ON	AH-E-4B	ON
AH-E-11A	ON	AH-E-11C	ON
RR-V25C	CLOSED	RR-V25C	CLOSED
BS-V1A	OPEN	BS-V1B	OPEN
CA-V10	CLOSED	CA-V1	CLOSED
CA-V4A	CLOSED	CA-V3	CLOSED
CA-V9	CLOSED	CA-V4B	CLOSED
RR-P-1B	ON	CA-V8	CLOSED
WDG-V199	CLOSED	CA-V6	CLOSED
WDL-V22	CLOSED	RR-P-1D	ON
WDL-V1126	CLOSED	WDG-V2	CLOSED
AH-D4092A & B	RECIRC	WDL-V1125	CLOSED
AH-D4092D & E	RECIRC	WDL-V271	CLOSED
ED-4098		AH-D4092A & B	RECIRC
		AH-D4092D & E	RECIRC
		ED-4098	RECIRC

TABLE B-1

ACUATION A		ACUATION B	
R.B. ISOLATION & COOLING GP. 3		R.B. ISOLATION & COOLING GP. 3	
Equipment	Position	Equipment	Position
IC-V2	CLOSED	IC-V3	CLOSED
IC-V5	CLOSED	IC-V4	CLOSED
MU-V2A	CLOSED	MU-V376	CLOSED
MU-V2B	CLOSED	MU-V18	CLOSED
MU-V377	CLOSED	MU-V25	CLOSED
NS-V72	CLOSED	NS-V100	CLOSED
NS-V81	CLOSED	NR-P-2B	ON
NR-P-2A	ON	RR-P-1C	ON
NR-V144A	OPEN	RR-V25D	CLOSED
RR-P-lA	ON	RR-V25E	CLOSED
RR-V25A	CLOSED	AH-C-8A	ON
RR-V25B	CLOSED	AH-E-11E	ON
IC-P-lA	OFF	AH-P-lB	ON
AH-C-8A	ON	AH-V2A	CLOSED
AH-E-11B	ON	AH-V2B	CLOSED
AH-E-11C	ON	AH-V3A	CLOSED
AH-P-lA	ON	AH-V3B	CLOSED
AH-1VA	CLOSED	AH-V6	CLOSED
AH-V1B	CLOSED	AH-V61	CLOSED
AH-V4A	CLOSED	AH-V61	CLOSED
AH-V4B	CLOSED	AH-V63	CLOSED
AH-V5	CLOSED	AH-V71	CLOSED
AH-V60	CLOSED	IC-P-lB	OFF
AH-V102	CLOSED		
AH-V72	CLOSED		
*BS-P-lA	ON	*BS-P-1B	ON

'If RB Pressure >30 psig.

Note 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (white).

Note 2: MU-P-lB will be running if normal power is available for the Actuation, for the pump that it is selected to backup. If normal power is lost, MU-P-lB will be running, if the pump that it is selected to backup fails to start or is inoperable.

Note 3: The NR pump in each header selected for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

Note 4: Normally NS-P-lA and 1B will start; however, NS-P-lC will start if either NS-P-lA or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

PANEL 8

Equipment	ES Position	Indication
DH-V7A(1)	Close	G
DH-V7B(1)	Close	G
NS-V83A	Open	R
NS-V83B	Open	R
NS-V215	Close	G
NS-V216	Close	G
CF-VIA	Open	R
CF-V1B	Open	R

(1) This valve may have to be opened for "piggy-back" operation. Once, opened, the Position/Indication becomes Open/R.

PANEL 15

Equipment	ES Position	Indication
DH-V6A(1)	Close	G
DH-V6B(1)	Close	G
MU-V378(2)	Open	R
MS-V4A(3)	Open	R
MS-V4B(3)	Open	R
MS-7A(3)	Open	R
MS-V7B(3)	Open	R

(1) This valve must be opened for sump-switchover. Once opened, the Position/Indication becomes OPEN/R.

(2) This valve should be closed at the operator's first chance. Once closed, the Position/Indication becomes CLOSE/W.

(3) These valves should be closed when the steam system is secured. Once closed, the Position/Indication becomes CLOSE/G.

PANEL 25

Equipment	ES Position	Indication
AH-E12A	Off	G
AH-E12B	Off	G
AH-E19A	Off	G
AH-E19B	Off	G

APPENDIX C

Checked Control Copy

EP/O/A/1800/4

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

LOSS OF REACTOR COOLANT

Considers the following cases:

Case A1: Excessive RC System Leakage - No Reactor Trip.

Case A2: Small Break -- Feedwater-RC Pumps-No Reactor Trip.

Case A3: Small Break -- Feedwater-RC Pumps-Reactor Trip.

Case A4: Small Break -- Feedwater-RC Pumps-Reactor Trip.

Case A5: Small Break -- Feedwater-No RC Pumps-Reactor Trip.

Case A6: Small Break -- Feedwater-No RC Pumps-RC Pressure Stabilizes at "Secondary Side Pressure.

Case A7: Small Break -- No Feedwater-No RC Pumps-Reactor Trip.

Case B: Rupture in Excess of Capability of Available High Pressure Injection pumps.

Case A4: Small Break--No Feedwater-RC Pumps-Reactor Trip

1.0 Symptoms

1.1 Excessive RCS makeup

1.2 Decreasing RCS pressure

1.3 Reactor trip

1.4 Decreasing Pressurizer level initially. May increase later.

1.5 RIA alarms

1.6 LDST level low or decreasing more than normal

1.7 ES actuation 1-2

1.8 Increasing Reactor Building Temperature and Pressure and Rx. Bldg. sump level

1.9 No feedwater flow and no S/G level

2.0 Immediate Action

2.1 Automatic

2.1.1 Reactor trip

2.1.2 Turbine trip

2.1.3 Possible ES actuation 1-2

2.2 Manual

NOTE: Any asterisk (*) parameters in the below sections shall be verified when step 3.1 of subsequent action **is** performed.

2.2.1 If ES Channels 1 & 2 have actuated because of a low pressure (*) condition in the RC system, IMMEDIATELY TRIP all RC pumps and refer Case A7 Section 2.2.

2.2.2 Verify automatic actions have occurred, if not, perform manually.

CAUTION: Do not override Automatic Actions of engineered safety features unless continued operation will result in unsafe plant conditions or will threaten reactor vessel integrity. (Refer to Enclosure 2).

2.2.3 Initiate ES 1-2 if it has not been actuated on ECCS signal.

CAUTION: If RC system pressure decreases below 1,600 psi(*) IMMEDIATELY TRIP all RC pumps and refer to Case A7 Section 2.2.

2.2.4 Check immediately for flow indication on both HPI emergency injection lines. If no flow is indicated in "B" loop, dispatch operator to open (2)HP-116 <u>within 10 minutes of ES actuation.</u>

2.2.4.1 For Unit 3, if no flow is indicated in "B" loop, open 3HP-409 <u>within 10 minutes of ES actuation.</u>

2.2.5 If no flow is indicated in "A" loop, dispatch operator to open (2)HP-26 <u>within 10 minutes of ES actuation.</u>

2.2.5.1 For Unit 3, if no flow is indicated in "A" loop, open 3HP-410 <u>within 10 minutes of ES actuation.</u>

CAUTION: If the HPI system has been actuated because of a low pressure condition, it must remain in operation until either:

2.2.5.2 Both LPI pumps are in operation and flowing at a rate in excess of 1000 gpm on header flow gauge (^)

each and the situation has been stable for 20 minutes,

OR

2.2.5.3 All hot and cold leg temperatures (%,) are at least 50 degrees below the saturation temperature for the existing RCS pressure on wide range pressure (*). If the 50 degrees subcooling by T_h indication (*) cannot be maintained after HPI cut-off, the HPI shall be reactivated (refer to Enclosure 1). The degree of subcooling beyond 50 degrees F and length of time HPI is in operation shall be limited by the pressure/temperature consideration for the vessel integrity (refer to Enclosure 2).

NOTE: If the HPI System has been activated and RC pumps operating, at least one RCP per loop shall remain operating.

2.2.6 Maintain maximum HPI flow. (^)

2.2.7 If pressure is increasing open RC-4 (Power Operated Relief Block) and RC-66 (Power Operated Relief Valve) to maintain forced cooling with the HPI system.

Note: If RC-66 is not operable, Pressurizer Code Reliefs will relieve overpressure and maintain force flow.

2.2.8 Monitor RCS T_{hot} (^)(if on scale) or incore thermocouples (^) (Display group #29) for indication of core outlet temperature stabilization. (T sat for 2,500 psig= 665°F).

2.2.8 Regain feedwater as soon as possible.

3.0 <u>Subsequent Actions</u>

3.1 Immediately on completion of necessary immediate manual action steps, alternate instrument channels shall be checked to confirm the key parameter readings that are marked with an asterisk (^), where alternate channels are available.

3.2 Once feedwater is available, commence feeding the OTSGs through the auxiliary feed nozzles and control level at ^ -25 inches on the startup range (^) and control OTSGs secondary side pressure on OTSG pressure gauge (*) at 'L1,000 psig using Turbine Bypass valves; if unavailable, the main steam relief valves.

3.3 Close the PORV, RC-66.

3.4 Regain RCS pressure control by energizing the pressurizer heaters and heating the pressurizer until the pressurizer temperature (^) indicates within the pressure temperature curve for saturation.

CAUTION: If pressurizer heaters are inoperable, control RCS pressure by throttling HP injection flow with (3)(2)HP-26 and (3)(2) HP-27.

3.4.1 If 3HP-26 and/or 3HP-27 fail, control RCS pressure by throttling 3HP-410 and/or 3HP-409 respectively.

3.5 Monitor RCS pressure carefully to ensure that the steam bubble is formed in the pressurizer

3.6 Place pressurizer heaters in automatic.

NOTE: RCS must be maintained subxooled by T_h indication (see Enclosure 1)

3.7 Borate the RC System for cold shutdown condition per OP/1103/15. Reactivity Balance Calculation.

3.8 Go to one (1) RC pump per loop operation. One (1) pump should be the pump that supplies pressurizer spray.

3.9 De-energize pressurizer heaters and maintain OTSG cooling by adjusting steam pressure using the Turbine Bypass valves or the manual steam dumps. Cooldown at 100 °F/hr. to achieve an RC pressure of 320 psig.

NOTE: Bypass ES low pressure injection and block core flood actuation at RC pressure of 700 psig.

NOTE: Plot RC pressure/RC temperature at 1/2 hour intervals on Enclosure 1 (Subcooled Curve).

3.10 Maintain RCS pressure at 320 psig and reduce RCS temperature to 240°F.

3.11 Stop one (1) RC pump.

3.12 Sample RC System for isotopic analysis and notify Superintendent of Operations of results prior to placing LPI in service.

3.13 Close (3)(2)LP-21 and (3)(2)LP-22.

3.14 Align and start "A" or "C" LPI pump in the decay heat removal mode (switchover for Unit 1&2) per OPs/1,2,3/A/1104/04. Establish > 1,000 gpm flow in the "A" header.

3.15 Stop the remaining RC pump.

3.16 Reduce RCS pressure to 100 psig by throttling HPI flow with (3)(2) HP-26 and (3)(2) HP-27. Maintain 50°F subcooling by throttling (3)(2) LPSW-251 and (3)(2) LPSW-252.

3.16.] If 3HP-26 and/or 3HP-27 fail, throttle HPI flow with 3HP-410 and/or 3HP-409 respectively.

3.17 Place LPI in normal decay heat removal mode per OP/1&2/A/1104/04 (Unit 1 & 2 only).

3.18 Open (3)(2)LP-22, close (3)(2) LP-21, start "B" LPI pump, open (3)(2) LP-18 and establish >1,000 gpm in "B" LPI train.

3.19 Secure HPI pump.

3.20 Shift LPI Pump "B" suction from the BWST to the Reactor Building Sump by opening (3)(2)LP-20 when sufficient NPSH is available.

NOTE: This is desirable to avoid unnecessary quantities of water in containment:

NOTE: To open 3LP-20, press and hold the LP-19 and LP-20 interlock bypass switch while opening 3LP-20.

3.21 Reduce RCS temperature per OPs/1,2,3/A/1104/04 using the decay heat removal coolers for long-term core cooling.

REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

CONTROL ROOM DESIGN AND PERFORMANCE

BY

Ronald M. Eytchison

October 1979
Washington, D.C.

TABLE OF CONTENTS

SUMMARY

There is evidence that the operators of Three Mile Island Unit 2 (TMI-2) were confused by equipment indications available to them on March 28, 1979. During the course of the accident which began that day, a number of malfunctions of control equipment occurred. This complicated the problem operators were facing or caused additional confusion. For this reason, the control room design was reviewed to evaluate both its adequacy in providing the necessary information to operators and the controls needed to shut down the plant and maintain it in a safe condition. Performance of the control room during the transient was assessed. Finally, industry efforts to improve control room design through human factors engineering were reviewed.

Historically, the design of the control room has been the responsibility of the architect engineer acting at the direction of the utility client. Nuclear Regulatory Commission (NRC) regulatory requirements address control room design only in a peripheral way. There is a statement in the Final Safety Analysis Report (FSAR) that suggests the control room is designed so that one person can operate the unit during normal steady-state conditions. There has not been standardization. Control rooms have generally evolved as certain designs tailored by the wishes of the client and influenced by precedent, designer preference, and nuclear steam system supplier recommendations.

The TMI-2 control room was the result of such an evolutionary process. It was designed so that it could be operated by a single person during steady-state conditions. Operating consoles were installed to include controls and indicators accessible and visible to the operator and to permit startup, normal operation, shutdown, and emergency operation of the nuclear power plant. Panels were arranged in a U-shaped pattern to provide ready accessibility to those control functions requiring the most frequent attention.

Review of the March 28 accident sequence indicates that the control room performed generally as designed. Operators were able to operate all critical equipment without undue difficulty. However, although the control room provided sufficient information to permit operators to cope with the transient, the confusion evident during the accident may have resulted in part from the control room layout and design or from the equipment malfunctions. The control room was evidently designed more for normal operation than for accident conditions.

The arrangement of controls and indicators for engineered safety features was not well thought out.

- o Emergency systems controls are not arranged in an orderly manner with all controls and process indications located in one section. For instance, the high pressure injection (HPI) throttle valves are operated from a front panel but the HPI flow indication is on a back panel and cannot be read from the throttle valve operating positions.

- o The TMI-2 control room alarm system provides audible and visual indication for most of the more than 1,500 plant alarm

conditions. The number of alarms is evidently to facilitate control of the entire plant by a single person during normal operating conditions.

- A single "acknowledge" button silences all of the alarms, making it likely operators could not comprehend the significance of all alarm conditions.

- The control room alarm annunciators are not arranged in a logical fashion. Annunciators associated with specific systems are distributed in a seemingly random fashion.

- Some audible alarms are associated with annunciators that are on the back sides of panels and cannot be seen by an operator standing in front of the related control panel.

- The existence of a large number of alarm conditions during normal operation tends to mask the alarm received during an emergency.

- During normal operation, indicator lights will be red, green, white, or amber, and it is not possible, at a glance, to detect an off-normal condition.

- The meaning of a given light color is not consistent among all of the panels in the control room.

- Computer aids for the analysis of system status were not utilized at TMI-2.

During the accident, the operators were initially confused by more than 100 alarms. They were misled by incorrect pilot-operated relief valve (PORV) position indications and ambiguous relief valve discharge line temperature indications. Shut indication for emergency feedwater block valves was not apparent, perhaps because of multicolored lights on the panels for normal conditions. There was no emergency feedwater flow indication available to alert the operators that the block valves were shut. Control of the condensate polisher bypass valves from the control room failed. Reactor coolant drain tank (RCDT) pressure indication, an important symptom of a stuck-open PORV, was not readily available to the operator.

Several operator aids not available during the accident might have assisted the operators as well as enhanced their ability later to analyze what happened. These include:

- improved computer capability for diagnostics and analysis of system status;

- instrumentation for the detection of inadequate core cooling using existing reactor instruments;

- display equipment consistent with the **physical process being described and graphic display where possible;**

- suppression of alarms during modes of operation when they are meaningless;

- a supervisory "command and control" panel on which key parameters would be displayed and which would be for the use of senior monitors; and

- a multichannel recorder which would record key parameters as well as have a voice channel.

Evaluation of the TMI-2 control room design and performance underscores the need for industry standards in this area. The design of the person-machine interface is recognized by industry to be less than adequate and efforts are being made to apply the principles of human engineering to future control room design. A standard, "IEEE Recommended Practice for the Design of Displays and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations," published in 1977, provides guidelines for an integrated and functional control room design. Other studies translate human engineering principles into practice in panel design and layout and assess the effects of human engineering on operator performance. Knowledge gained in this work may be applied to future design and, where possible, backfitted into existing control rooms.

INTRODUCTION

Nuclear power plants are relatively complex facilities operated by few persons. The number of systems and controls which interact with and require manipulation by the operators is quite large. Effectiveness of operator action is directly related to the clarity of process information and the availability and orientation of controls, indicators, and alarms. Ambiguous information during periods of stress can add confusion to an already difficult situation. Clearly, the control room of a nuclear generating station has a significant role during normal plant operation and is even more important during accident conditions in permitting operators to place the plant in a safe condition.

Because there was evidence that confusion of operating personnel may have contributed to the absence of proper response to the accident conditions at TMI-2, it is important that the consistency of the TMI-2 design with good practice, industry standards, regulatory positions, and general design criteria be evaluated. Additionally, assessment of the control room performance vis-a-vis the expected response is needed.

The control room in TMI-2 was inspected by members of the Commission staff during visits to the plant on June 11 and June 26, 1979. Control room guidelines currently in effect in the NRC Standard Review Plan and IEEE 566-1977 were also reviewed, as well as a number of industrial standards discussed in Appendix A.

ANALYSIS

GENERAL

The purpose of a nuclear station control room, as described in Section 7.5 of the TMI-2 FSAR is to provide in one room the control stations, switches, controllers, and indications necessary to start up, operate, and shut down the nuclear unit. Control functions necessary to maintain safe conditions after a loss-of-coolant accident (LOCA) are initiated from the control room. Controls for certain auxiliary systems are located at remote stations when the system controlled does not involve power generation control or emergency functions.

Historically, the design of control rooms has been the responsibility of the architect engineer acting at the direction of the utility client. Regulatory requirements address control room design only in a peripheral fashion (via general design criteria), and industry standards, which relate to control room layout, are at best qualitative in nature. Thus, control rooms have generally evolved as custom designs tailored by the needs of the client. The history of development of the TMI control room design and its construction is described and analyzed in another staff report.1/

NUCLEAR REGULATORY COMMISSION REQUIREMENTS AND INDUSTRY GUIDANCE

Review of regulations which might relate to nuclear power plant control rooms determines that there are no regulatory requirements other than those in Appendix A to 10 CFR 50 (general design criteria).

The NRC's Standard Review Plan (SRP), which was issued in 1975 and codified practices in effect before that date, does not specifically address control room layout. Section 7.0 of the SRP relates to review of instrumentation and control systems, reactor trip systems, engineered safety features systems, systems required for safe shutdown, and the like, but it does not treat the control room as a whole. The SRP provides for an NRC review of the control room during a site visit near the end of the operating license review period. However, the SRP does not require a specific review of the control room design layout because the control room is not specifically categorized as a safety-related item. Therefore, the impact of the design on overall plant safety is not addressed by the license applicant in the Final Safety Analysis Report (FSAR) or by the NRC in its license application finding.

The general design criteria of 10 CFR 50, Appendix A, include in Criterion 13 (Instrumentation and Control) and Criterion 19 (Control Room) some brief minimum requirements for design criteria but contain nothing concerning the layout of control room designs.

In the absence of stringent criteria relative to control room design, other factors do have some influence on the design. These include precedent, designer preference, operator preference, and nuclear steam supply system supplier recommendations.

DESCRIPTION OF THE TMI-2 CONTROL ROOM

General Layout

The control room at TMI-2 was designed so that one person can operate the unit during normal steady-state conditions. It is assumed that other operators will be available to assist the control room operators during other than normal operating conditions. The control room is arranged to include the operating consoles, which house frequently used normal controls and indicators, as well as startup and emergency controls and indication. Controls and indication are intended to be logically arranged so that controls are accessible and indicators readily visible to the operator. Remaining indicators, infrequently used control switches, and all annunciators are mounted on vertical panelboards behind the consoles and are to be readily visible to the operator. Layout of the control room is shown in Figure 1.

Information Display and Control Functions

Information considered by the designer to be necessary for routine monitoring of the unit is displayed on the control room consoles or on panel boards behind the consoles. Information display and control equipment frequently employed, startup controls, protective equipment needed in case of an emergency, and the computer panel are mounted on the consoles.

Recorders and radiation monitoring equipment, infrequently used control switches, remaining indicators, temperature recorders, annunciators, and reactor building isolation valve position indicators are mounted on the vertical panels.

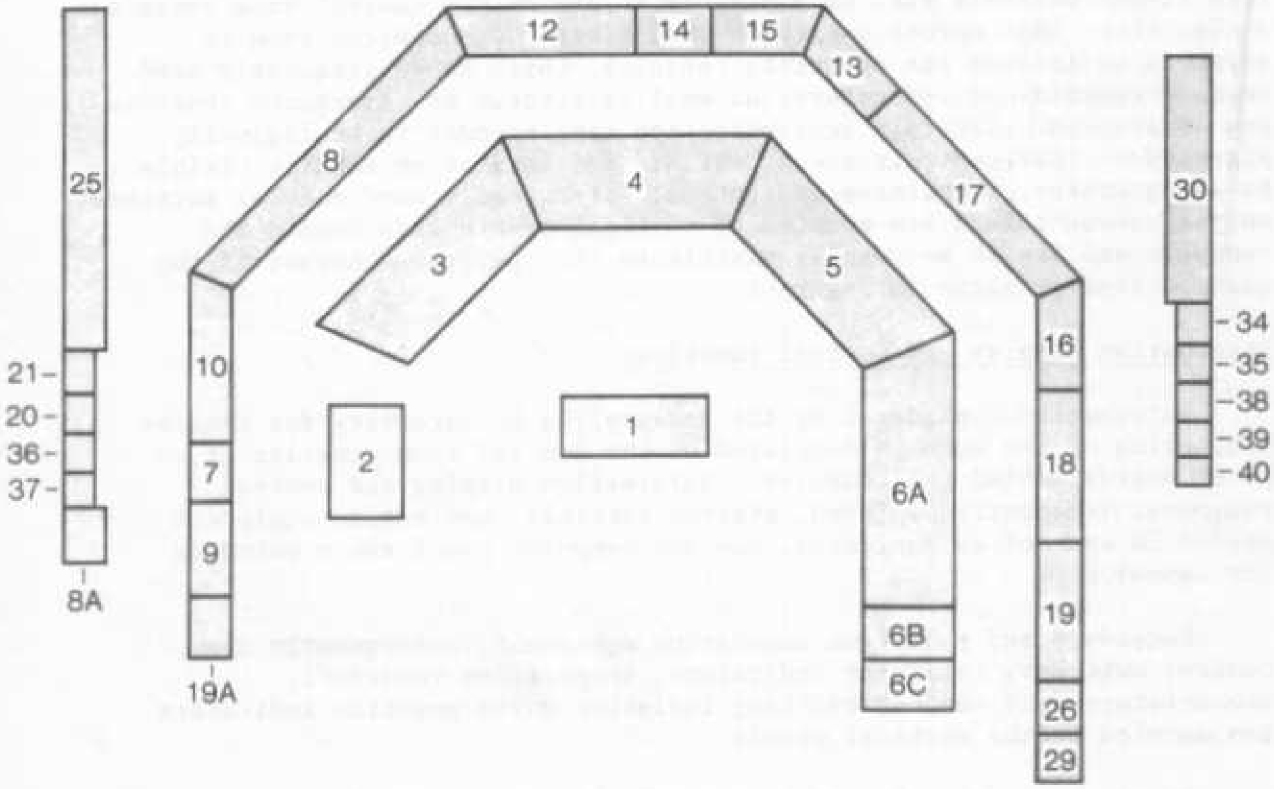
As can be seen on Figure 1, the consoles are arranged in a U-shaped pattern. The vertical panels follow the same panels behind the consoles, with a passage aisle separating the consoles and the vertical panels. The operator's desk is located in front of the U-shaped console and panel arrangement. A computer is available in the control room on the computer console for alarm monitoring, performance monitoring, and data logging. On-demand printout is available to the operator in addition to periodic logging of the unit variables.

Consoles and Panel Layout

The designer intended, as expressed in the FSAR, that control consoles and panels be arranged to provide ready accessibility to those control functions requiring the most frequent attention, to group control and readout devices in relation to their function, and to maintain physical separation between redundant safety-related systems control and indication. The panels contain the necessary controls and indication for monitoring conditions in the reactor, reactor coolant systems, containment, and safety-related process system for all plant operating conditions.

A brief description of each of the panels in the control room follows. Panel numbers refer to those shown in Figure 1. Panels 2

FIGURE 1: Control Room Panel Layout



Source: NSAC, "Analysis of Three Mile Island-Unit 2 Accident," NSAC/Electric Power Research Institute, NSAC-1, July 1979.

through 6 are console-type panels and the remainder are vertical. Panel 2 is the computer console. Panel 3 includes the reactor coolant makeup and justification system and the control equipment related to the safety features actuation system. Panels 4, 5, and 6 contain the controllers, recorders, and indications necessary for control and supervision of the reactor power output, feedwater, condensate, steam generators, and turbine generator. Panel 7 indicates the existence of a fire in the unit and the automatic steps being taken to control it. Panel 8 contains the annunciators and indication for status of the various nuclear and conventional cooling systems of the unit. Panel 10 records temperatures of major equipment, reactor vent valves, control rod drives, and self-powered neutron detector tubes -- each temperature monitored is alarmed if the temperature exceeds a present limit. Panel 12 contains station radiation monitoring equipment and recorders; included here is equipment required to annunciate and indicate the status of equipment and interlocks intended to prevent any release to the environment that exceeds present limits. Panel 13 displays the status of the engineered safety features panel. Panel 14 contains information on each individual control rod and includes rod position, fault lights, and inserted and withdrawn limit lights. Panel 15 is a graphic panel that shows the position of all reactor building isolation valves.

Computer 2/

TMI-2 uses a Bailey 955 computer that is linked to a smaller NOVA computer to form an integral system. The latter computer is principally for monitoring the balance-of-plant parameters. The main purpose of the computer system is to monitor continually about 3,000 plant parameters and display the values of the parameters or calculations based on the values. Parameter input signals are either analog or digital.

The computer monitors 960 digital and 80 analog inputs each second. Depending on its importance, an analog parameter may be scanned on 1-, 5-, 15-, 30-, or 60-second intervals.

The computer has two output modes for the points it scans -- an alarm printer and a utility printer. A small cathode ray tube display duplicates the output of the printer. The alarm printer automatically prints an alarm message when the parameter has gone into an alarm condition. Alarm inputs are stored by the computer in an alarm backup file or buffer until they are printed; this buffer can store up to 1,365 alarm inputs before it is filled. The alarm printer can print only one alarm every 4.2 seconds. If alarms are occurring at a faster rate, the printer gets further and further behind. At one time during the accident the alarm printer was at least 161 minutes behind. If the number of backlogged alarms exceeds 1,365, additional alarms will be neither stored nor printed.

The utility printer will provide the value or condition of any monitored parameter when requested.

EVALUATION

One cannot say that the control room of TMI-2 led directly to the onset of the transient or follow-up events. Nevertheless, although the control room apparently should have provided the operators with sufficient information to permit them to cope with the transient and its aftermath, the operator confusion evident during the accident may have resulted in part from the control room layout and design or from control room equipment malfunctions. Defects in the design and control room equipment malfunctions are discussed below to assess their significance to the accident.

Arrangement

The general arrangement of the control room contains several deficiencies that detract from the intended function.

The U-shaped front panel arrangement is to provide the operator with quick access to all major plant controls with a minimum of physical movement. This panel is supplemented by a parallel rear panel which contains important indications and controls needed to control the plant during emergencies. A separation of electrical distribution, steam (secondary) plant control, and nuclear steam supply sections of the plant control is consistent with industry practice. However, the lack of ready access to this back panel from the front is evident in Figure 1.

Many indicators on the back panel are so far away from the operator they cannot be read, and the standby operator must walk to the back panel to read them. Moreover, many indicators and alarm annunciators are located on the back side of the rear panel where they cannot be seen at all from the operating area. An operator located at the center of the front panel would have to walk more than 70 feet to gain access to a control or read an indicator only about 6 feet away, even though access to the back panel is necessary during upset conditions. For example, the flow indication from high pressure injection is located on panel 8 and is not readable from panel 4 from which the high pressure injection throttle valves are operated. The large size of the TMI-2 control room and the lack of a passageway between panels 5 and 6A means that during upset conditions the operators on a normal shift can have considerable difficulty performing and coordinating all the necessary recovery functions at both panels.

The TMI-2 plant has approximately 60 specific systems that perform various functions necessary for the generation of electricity. These systems interrelate and the function of one may affect the operation of the plant as a whole. It is necessary that the operator comprehend the individual systems that make up the whole, particularly during an upset situation. Having information available on a systems basis permits the operator to review quickly valve lineups or equipment status. It is important that the arrangement of controls, indication, and switches be such that systems status is clear, centralized, and unambiguous. This can best be accomplished by providing component information in an organized, systematic fashion.

Review of the TMI-2 control room indicates that for normal operating conditions equipment and components status is adequately displayed on a systems basis. The internal plant electrical distribution system is clearly displayed with a "mimic" bus indicating interconnecting busses and circuit breakers as they functionally relate to each other. The makeup and letdown system within the auxiliary building is also presented in a "mimic" fashion, but less than the entire system is presented on the front panel.

Engineered safety features systems are not grouped by system but are separated with pump controls on the front panel and various valve controls on the rear panel. Process instrumentation for the engineered safety features system is separate from the pump and valve controls. Emergency systems controls are not arranged in an orderly manner with all controls and process indication located in one section. Controls for containment systems that interconnect with components inside the reactor building are centrally located on one panel which displays the information needed to inform the operator of the status of containment penetrations. Valves associated with engineered safety features (ESF) are also presented on a separate features panel that indicates valve position and pump or fan motor breaker status. The combination of a section for all containment penetration valves and a centralized indication panel only for engineered safety features component status, with no component system controls and indication, results in a confusing design.

Alarms

The TMI-2 control room alarm system provides audible and visual indication for most of the more than 1,500 plant alarm conditions. Such a large number of alarms is evidently to facilitate control of the entire plant by a single person during normal operating conditions. During an emergency situation, however, the alarm systems confuse, rather than help, operators as they try to analyze the casualty. The loss of main feedwater, the turbine trip, and the reactor trip on March 28 resulted almost simultaneously in more than 100 alarms. Assimilation of this amount of information while many complex events were occurring was not possible. A single "acknowledge" button silenced all of the alarms, making it even more likely that the operators were unable to comprehend the significance of all the alarm conditions.

Further complicating the situation was the failure of the designer to arrange the control room alarm annunciation in a logical fashion. Annunciators associated with specific systems are distributed in what appears to be a random fashion. There are no system groupings or placement of annunciators directly in front of the related control panel. Some audible alarms are associated with annunciators that are on the back sides of panels and that cannot be seen from the front of the related control panel. Alarm lights for rather trivial conditions are intermixed with important alarm lights for safety systems. For instance, turbine building elevator trouble, auxiliary building elevator trouble, and reactor building elevator trouble lights are in the same annunciator group as alarms associated with loss of coolant. Significant alarm conditions are not distinguishable by a special color, such as red.

Finally, the existence of a large number of alarm conditions during normal operation tends to mask the alarm received during an emergency. A shift supervisor remarked that the smallest number of alarms he had ever observed in TMI-2 before the March 28 accident was 52. 3/ Continuous alarms due to alarm malfunction or because of long-term conditions that cannot be corrected cause confusion.

Indicator Lights

The control consoles at TMI-2 include hundreds of lights indicating the status of valves (open-shut), motors (on-off), and circuit breakers (open-shut). During normal operation these lights will be red, green, white, or amber, and it is not possible, at a glance, to detect an off-normal condition easily. Again, during an emergency situation the existence of a valve, motor, or breaker in other than its expected condition is not immediately apparent. In addition, the meaning of a given color is not consistent between all of the panels in the control room.

Computer Aids

Computer aids for the analysis of system status were not utilized at TMI-2. These could play an important role in handling emergency situations.

PERFORMANCE OF THE CONTROL ROOM DURING THE ACCIDENT

Review of the performance of the TMI-2 control room on March 28, 1979, indicates that it fulfilled the requirements of the FSAR; that is, it provided the control stations, switches, controllers, and indications necessary to shut down the plant. Information indicating that a LOCA was in progress was available to and known by the operators. Control functions necessary to maintain safe conditions after a LOCA were available to the operators. Nevertheless, a number of failures and inadequacies contributed to operator confusion and made accident mitigation more difficult. The large number of alarms received concurrent with the loss of feedwater, the turbine trip, and the reactor trip caused a great deal of confusion. In the view of one control room operator the alarm panel was useless. 4/ Design problems with the alarm system, foreseen by the second control room operator nearly a year before the accident,5/ caused difficulty on the morning of March 28.

The operators were misled by position indication of the PORV as displayed on panel 4, the reactor console. Open or shut indication of this valve was indirect rather than positive. Although the PORV was stuck open following the initial pressure transient control, panel lights indicated that the valve was closed because the signal was derived from a sensor of solenoid energization rather than a sensor of valve position itself. The operators relied heavily on this indication rather than considering other symptoms of a stuck-open PORV.

Perhaps related was misunderstanding of the significance of the control room indication for relief and safety valve discharge piping temperatures. Contributing to this was the ambiguity of indication that if one of these three valves leaks or opens, all three will register a high temperature making it difficult to determine which valve passed the steam. Although the operators requested these temperatures several times from the computer in the period that the PORV was open, they were not alerted to the fact that the valve was open by the temperature indications.

The reactor coolant drain tank (RCDT) pressure indication, an important symptom of a stuck-open PORV, was not readily available to the operator. Rather, it was displayed on the back side of panel 10. Because so much attention was required at the control panels it was impractical to monitor RCDT parameters frequently.

At the start of the transient and for about 8 minutes thereafter the emergency feedwater block valves were shut, causing the steam generators to boil dry. Although indicating lights showed the valves to be shut, they were not of a distinctive color to alert the operators that the valves were out of their proper position.

Failure of the operators to detect the lack of emergency feedwater flow to the steam generators was an important factor affecting heat removal capability early in the TMI-2 accident. The availability of emergency feedwater flow indication might have helped the operator quickly determine that the pump discharge valves were shut. This flow indication was not available.

Another factor that contributed to confusion was difficulty in restoring the condensate system to operation. Compounding this problem was the fact that the electrical operation of the condensate polisher bypass valve could not be carried out from the control room. Nearly an hour was required to open the bypass valve manually.

During a period of about 73 minutes early in the accident, the alarm printer was unavailable because of a paper jam. However, this was only a minor operational inconvenience because the printer was running significantly behind real time due to the large number of alarms received.

It is likely then that although a number of malfunctions occurred in the TMI control room none was of crucial importance. Overall, the instruments in the control room provided vital information to the operators, who failed to recognize its significance. The control room also provided adequate controls to mitigate the accident.

AIDS THAT MIGHT HAVE ASSISTED THE OPERATOR

In addition to aids to the operator whose need is obvious from the previous discussion -- fewer alarms; passageways between panels; a smaller, more compact control room; positive position indications for the PORV; emergency feedwater flow indication; and so on -- there are other features which, if incorporated, might facilitate operation of the plant. These are:

- o improved computer aids for the analysis of system status;
- o instrumentation for the detection of inadequate core cooling using existing reactor instrumentation (flow, temperature, pressure, power);

- display equipment that would facilitate operator understanding, e.g., physical differentiation of data presented, data displays consistent with the physical process being described, and graphic displays whenever possible;

- suppression of alarms during operation when they are meaningless;

- a supervisory "command and control" panel on which key parameters would be displayed and which would be used by senior monitors (this would provide overall plant status without the monitors having to approach the consoles or interfere with the operators); and

- a multichannel recorder that would record key parameters as well as have a voice channel (the recorder could be programmed to start on initiation of an upset event such as a turbine trip or reactor trip and could be used to assess plant and operator performance).

Evaluation of the design and layout of the TMI-2 control room underlines the need for industry standards in this area. There is evidence that the industry recognizes the design of the person-machine interface to be less than adequate and is trying to apply the principles of human engineering to future control room design. An important standard published in 1977 that provides guidelines to help engineers make decisions is "IEEE Recommended Practice for the Design of Displays and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations" (IEEE Std. 566-1977). Details of IEEE 566-1977 are given in Appendix A.

This document recommends that three basic steps be followed in sequence to provide an integrated and functional control room design. Before beginning the detailed design of the control room the architect engineer should, in consultation with the utility:

- Develop the design bases for control and display facilities.

- Develop a usage analysis to establish and document a systematic method for assignment of control/ indication locations.

- Consider human engineering factors. Control and display features of the design should be such that indication can be easily used as a tool for analysis of the status of specific plant systems and control features.

Appendix A also includes a discussion of other industry documents relating human factors research to the design of reactor control systems and control room layout. Efforts of major nuclear steam supply suppliers to develop design concepts based on human engineering principles are summarized. A study performed by Lockheed for the Electric Power Research Institute evaluated control room designs of several currently operating nuclear power plants with respect to translation of human engineering principles into practice in panel design and layout. Another study, entitled "Human Engineering of Nuclear Power Plant Control Rooms and Its

Effect on Operator Performance," prepared in 1977 for the NRC by the Aero Space Corporation provides an assessment of the effects of human engineering on operator performance. Finally, Appendix A discusses an analysis performed by Sandia Laboratories, "Preliminary Human Factors Analysis of Zion Nuclear Power Plant." This was developed to provide the basis for incorporation of human factors into the design of future plants.

FINDINGS

1. At the time the TMI-2 control room was designed there were neither regulatory requirements nor industry standards relating to control room design and layout. In the absence of stringent criteria, design was the result of precedent, designer preference, operator preference, and nuclear steam supply supplier recommendations.

2. The TMI-2 control room was designed so that it could be operated by one person. This is desirable for normal operation and undesirable for emergency conditions.

3. The general layout, information display, control functions, consoles, and panel layout provided the operators with sufficient information and controls to permit them to cope with the March 28, 1979, transient and its aftermath.

4. The TMI-2 control room did not lead directly to the onset of the transient or the follow-on events.

5. Operator confusion, which was evident during the accident, was increased by the control room layout and design and by specific control room malfunctions.

NOTES

1/ See the report of the Office of Chief Counsel, "The Role of the Managing Utility and Its Suppliers," President's Commission on the Accident at Three Mile Island, October 1979.

2/ Adapted from Appendix PDS to Analysis of Three Mile Island Accident (NSAC-1), Nuclear Safety Analysis Center, Palo Alto, California, 1979.

3/ W. Zewe interview on June 21, 1979.

4/ Faust in May 30, 1979, public hearing.

5/ Frederick, as quoted in staff report on "The Role of the Managing Utility."

APPENDIX A

REGULATORY AND INDUSTRY GUIDANCE
REGARDING CONTROL ROOM DESIGN

The Standard Review Plan (SRP) does not specifically address control room layout in Section 7.0 which relates to review of instrumentation and control systems, reactor trip system, control of engineered safety features systems and auxiliary systems, systems required for safe shutdown, safety-related display instrumentation, and all other instrumentation systems not required for safety. Appendix 7B of Chapter 7.0 does provide guidance relative to review of the control room during a site visit performed by members of the NRC's Instrumentation and Control Systems Branch near the end of the operating license review effort. However, the guidance in Appendix 7B does not require a finding by NRC staff on the adequacy of the overall design. Although reference is made to the control room design in Section 13.1.1, specifically with regard to utility efforts in "Development of human engineering design objectives and design phase review of proposed control room layouts," a utility effort which "should be substantially accomplished before preoperational testing begins and generally before submittal of the final safety analysis report," no specific requirement for a finding relative to overall design adequacy is required. Consequently, it can be concluded that the SRP does not require a specific review of control room design layout because the control room is not specifically categorized as a safety-related item, although it is designated a vital area for purposes of other NRC reviews (fire protection and security). Therefore the impact of the design on overall plant safety is not addressed by the utility applicant in the FSAR, nor by the NRC in its finding on the overall application.

The General Design Criteria of 10 CFR 50, Appendix A, address criteria for the layout of control room designs.

In the absence of stringent criteria relative to control room design, utility applicants rely on other means of design. Such means include precedent: designing a new control room to look like a previous layout. NRC staff references an industry standard, IEEE 566-1977, when asked about guidance regarding control room design. Information provided in this standard is aimed at providing "guidelines to aid designers in making decisions" rather than rigid mandatory standards. Selection of display and control equipment and operating requirements for individual plant systems are established to a great extent by systems designers. The integration of information, control, and operating requirements is a task now performed by the architect engineer in development of control room designs.

IEEE 566-1977 recommends that three basic steps be followed in sequence to provide for an integrated and functional control room design. These are:

- development of design bases;
- establishment of a systematic method for assignment of types and location of controls and displays (usage analysis); and

- consideration of human engineering factors.

Before beginning the detailed design of the control room, the architect engineer should, in consultation with the utility, develop the design bases for control and display facilities. The design basis should include a list of operating modes, number of operators and designation of their responsibilities, and definition of the functional areas into which the control room is organized (for normal, emergency, and supporting operations). The basis for grouping control and display devices should be developed. The maximum number of display devices and alarms that can be active at the same time should be established with the objective of preventing the operator from becoming overloaded with various means of indication competing for attention. A list of safety-related display and control equipment required by Safety Analysis Reports, regulatory requirements, or industry standards should be developed. Additional requirements such as those dictated by the utility should be noted. Finally, the design basis should take into account the sequence of events for design basis accidents, use of remote panels, and the "anthropometric relationship to be used in design of control boards."

The development of a usage analysis is the second step in control room design evaluation. The purpose of this analysis is to establish and document a systematic method for assignment of control/indication locations. IEEE 566-1977 recommends the following identification scheme for each function by its usage characteristics:

- priority and importance of information or action
- plant systems
- operating modes
- frequency of use
- response time
- safety classification
- grouping of displays and controls in a functional area.

The second step in the usage analysis is to develop a set of criteria for device assignment and location based on usage characteristics already determined. After completion of the second part of the usage analysis, the criteria should be checked against applicable design bases to ensure consistency.

Functional or human engineering principles that form the third part of the methodology consider the operator to be a fundamental part of an integrated system necessary for safe and efficient plant operation. Control and display features of the design should be such that indication can be easily used as a tool for analysis of the status of specific plant systems and control features. Control functions and associated displays should be located so as to facilitate operator understanding, decision-making, and subsequent action.

Display features should be accessible, easily read, and comprehensible, and alert the operator to abnormal, unsafe conditions in the plant or to significant changes in plant status. The need for specific information to be displayed should be based on the following priorities:

- consequences of the operator's not taking action;
- importance of the information to the operator in determining plant status; and
- the degree of automation used in control system design.

In addition, use of alternative techniques such as display by exception should be considered at this point.

With regard to operator comprehension, display equipment should provide means to facilitate operator understanding. Principles to be considered in terms of operator comprehension are:

- physical differentiation of data presented (attention to color coding, size, and shape);
- formats for data display that are consistent with physical process being described, for example, use of a vertical bar indication for level; and
- graphic displays that would enhance comprehension.

Indication should be employed to the greatest extent possible on development of a true abnormal condition within the plant. Alarms should be suppressed during modes of operation in which they are meaningless. During changes in operating modes, information necessary for that mode only should be made available for operator use.

Finally, facility technical specifications define operational limits. These limits are placed on systems monitored by the operator; provisions should be made in terms of indication to facilitate operator cognizance of the limits.

General Electric Corporation (GE) had developed an advanced modular control room design for installation in their boiling water reactor plants. The GE Nuclenet is a modern control room complex for which complete factory fabrication and checkout is possible prior to delivery to the customer. "Functions required for plant operation and control devices which the operator routinely operates are integrated and centralized on the control board." The basic philosophy behind the design is to "optimize the quantity of data presented to the operator, the format of data presented, and the number of display devices which must be monitored and understood. While the computer has been used extensively to format and drive the CRT display system in the Nuclenet control center, it has not been employed as an integral part of the control system per se. Additionally, in deference to current licensing requirements, a standby information panel is included which contains redundant handwired displays of critical items for use in the event of computer failure. GE has stated that display formats, control locations, and system design in general have been the result of extensive human engineering analyses."

Babcock & Wilcox has also developed an Advanced Control Center based on human engineering principles. This control center, like the GE Nuclenet, "employs CRT displays which are available to assist the operator during abnormal conditions but which are not interactive with controls nor required for abnormal operation. Whereas the Nuclenet concentrated its CRT displays in its master console, the B&W design has CRT displays distributed to some of the auxilary and safety center benchboards."

A third firm, Combustion-Engineering, has developed an advanced computer-based design concept. Basic design philosophy, as for other advanced concepts, attempts "to organize the vast amount of information available to the operator into a format which can be readily comprehended. The objective is to assist the operator in correctly interpreting plant status and thereby increase probability of timely and appropriate response to abnormal conditions. This objective is achievable with a careful application of human engineering techniques in the overall design of the control center with particular emphasis on the engineering of the displays available to the operator."

The advanced design concepts referenced rely in principle upon application of human engineering techniques. Use of these techniques in development of recent designs has been the subject of attention in recent literature. Human engineering is basically an "interdisciplinary specialty concerned with influencing the design of equipment systems facilities and operational environments to promote safe, efficient, and reliable operator performance."

A 1976 study performed by Lockheed Missiles and Space Company, Inc., for Electric Power Research Institute, evaluated control room designs of a sample of currently operating nuclear power facilities relative to translation of human engineering principles into practice in panel design and layout. The basic conclusion of the Lockheed study is that additional research is required to determine how best to optimize control room design. The study states:

> As a first priority a detailed set of applicable human factors standards must be developed and industry-wide acceptance should be promoted. These standards would serve to provide design and evaluation criteria. . .such standards will stimulate a uniform and systematic concern for human factors in design considerations.
>
> In addition to a comprehensive set of standards a need is perceived for human factors engineering design guides specific to the needs of the nuclear power industry. Such guides should first address realistic and cost-effective methods for upgrading present operational control rooms and second, guide the development of future control room designs.

Additional specific recommendations relate to development of an extensive research program within the nuclear industry to encompass the following topics:

- o Major selected control board panels should be reconfigured in terms of human factors engineering criteria and an evaluation of operator performance should be conducted to compare existing panels with human engineered layouts.

- A definite comparative evaluation should be conducted to determine the advantages of conventional versus advanced, computer-based, control board concepts.

- Candidate warning-annunciator approaches need to be defined and evaluated to provide improved operator-alerting systems and diagnostic aides.

- An investigation should be conducted of decision-aiding techniques, such as fault trees, that might help the operator to identify, integrate, and assess presently scattered bits of information.

- The human factors of the auxiliary operator's role, his interaction with the control room operator, and the person-machine interfaces associated with the auxiliary operator's task should be reviewed.

- Nuclear power plant communication systems should be thoroughly investigated to develop recommendations for facilitating normal and emergency control room operator contacts with other plant personnel.

- The ease of maintenance of nuclear power plant hardware systems, the adequacy of maintenance instructions and procedures, and the training and selection methods for maintenance personnel should be evaluated.

Another study, entitled "Human Engineering of Nuclear Power Plant Control Rooms and Its Effect on Operator Performance," prepared in February 1977 by the Aerospace Corporation for the NRC, provides a more comprehensive assessment of the effects of human engineering on operator performance in the control room. This study:

> was focused primarily in control rooms and control system design influences on the operator. Some observations on the influences of operator characteristics and job performance guides (operating procedures) were also developed.

The study focused on three groups of factors that influence operator performance in fulfilling control room functions, namely:

- control room and control system design;

- operator characteristics; and

- job performance guides (operating procedures).

Specific recommendations resulting from this study include:

- development of NRC regulatory guide for human engineering of control rooms to provide specific and detailed direction to control room designers;

- development of a licensee event report-type data base on personnel errors to establish an operator-reliability data base (the study would support determination of specific areas of weakness in power plant design which contribute to operator error); and

- review of simulation of transients and accidents currently utilized in operator training programs (this study would determine whether existing simulators have the capability to provide adequate training in emergency condition response).

Another human factor analysis performed by Sandia Laboratories, "Preliminary Human Factors Analysis of Zion Nuclear Power Plant," was developed to provide a basis for incorporation of human factors into the design of future plants. Commonwealth Edison's Zion Nuclear Power Plant was evaluated.

Human factors observed **in** this study fall into three categories:

- human engineering design deficiencies;

- deficiencies in training/practice; and

- deficiencies in operating instructions.

Conclusions reflect opportunities for improvement:

- Human factors techniques can be used to determine the degree to which design parameters, provisions for training and practice, and operating procedures concur with reliable human operations.

- The design of control room panels. . .deviates from accepted human engineering standards and increases the probability of human errors that could lead to the unavailability of safeguard systems. Design of future plants should incorporate human factors at all phases of design. . . and should include all person-machine interfaces such as maintenance, calibration, and control room operation.

- Material improvement in human performance. . . could be achieved by means of some relatively minor and inexpensive changes to plant equipment. . ., increasing provisions for practice of emergencies, and changing the format and content of written procedures.

- Valuable data on human performance can be collected during all phases of nuclear plant scheduled tasks (maintenance, calibration, periodic tests, etc.) and on control room tasks. . . .

- Industry-wide standards covering all aspects of human reliability. . . could serve to improve materially the impact of human performance on system availability and safety.

REFERENCES

1. Final Safety Analysis Report, Three Mile Island Nuclear Station Unit 2, Metropolitan Edison, Jersey Central Power and Light and Pennsylvania Electric Company, April 1974.

2. Code of Federal Regulations, Title 10, Appendix A, as revised January 1979.

3. Burns and Roe Conference Notes No. 235, dated Dec. 26, 1968.

4. Burns and Roe Conference Notes No. 74, dated Feb. 6, 1968.

5. Burns and Roe telephone conversation, S. Gottilla with J. Bartman, Metropolitan Edison, dated Jan. 13, 1969.

6. Burns and Roe telephone conversation, S. Gottilla with R. Burns, Jersey Central Power and Light, and V. Thomas, General Public Utilities, dated Jan. 15, 1969.

7. Burns and Roe Conference Notes No. 273, dated March 18, 1969.

8. Burns and Roe telephone conversation, S. Gottilla with J. Bacafer, Metropolitan Edison, dated March 20, 1969.

9. Letter, J. Miller, Metropolitan Edison, to P. Nardone, Burns and Roe, dated June 26, 1969.

10. General Public Utilities memorandum, R. Williams to G. Bierman and J. Neely, dated March 14, 1969.

11. Burns and Roe memorandum, S. Gottilla to E. Gahan, dated Dec. 27, 1968.

12. NUREG 75/087, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Nuclear Regulatory Commission, revised, March 1979.

13. IEEE Std. 566-1977, "IEEE Recommended Practice for the Design of Displays and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations," Nuclear Power Engineering Committee of the IEEE Engineering Committee, 1977.

14. U.S. Nuclear Regulatory Commission Meeting Summary, F. Ashe to F. Rosa, dated Sept. 8, 1977.

15. ART-77(2815)-l, The Aerospace Coporation for the U.S. Nuclear Regulatory Commission, "Human Engineering of Nuclear Power Plant Control Rooms and Its Effects on Operator Performance," February 1977.

16. Myers, J. and Schulty, T., "A Computer Based Design of a Power Plant Control Center," IEEE Transactions on Nuclear Science, Vol. NS-22, February 1975.

17. EPRI NP-308-SY, Lockheed Missiles and Space Company for Electric Power Research Institute, "Human Factors Review of Nuclear Power Plant Control Room Design," November 1976.

18. NUREG 76-6503 (SAND 76-0324), Sandia Laboratories, "Preliminary Human Factors Analysis of Zion Nuclear Power Plant," October 1975.

0 U. S. GOVERNMENT PRINTING OFFICE :1980 307-309/6795

Staff Reports To

THE PRESIDENT'S COMMISSION ON
THE ACCIDENT AT
THREE MILE ISLAND

The Nuclear Regulatory Commission, Report of the Office of Chief Counsel

The Role of the Managing Utility and Its Suppliers, Report of the Office of Chief Counsel

Emergency Preparedness, Emergency Response, Reports of the Office of Chief Counsel

Reports of the Technical Assessment Task Force, Vol. I
- "Technical Staff Analysis Reports Summary"
- "Summary Sequence of Events"

Reports of the Technical Assessment Task Force, Vol. II
- "Chemistry"
- "Thermal Hydraulics"
- "Core Damage"
- "WASH 1400 -- Reactor Safety Study"
- "Alternative Event Sequences"

Reports of the Technical Assessment Task Force, Vol. III
- "Selection, Training, Qualification, and Licensing of Three Mile Island Reactor Operating Personnel"
- "Technical Assessment of Operating, Abnormal, and Emergency Procedures"
- "Control Room Design and Performance"

Reports of the Technical Assessment Task Force, Vol. IV
- "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"

Reports of the Public Health and Safety Task Force
- "Public Health and Safety Summary"
- "Health Physics and Dosimetry"
- "Radiation Health Effects"
- "Behavioral Effects"
- "Public Health and Epidemiology"

Report of the Emergency Preparedness and Response Task Force

Report of the Public's Right to Information Task Force