COMMITTEE ON INTERIOR AND INSULAR AFFAIRS
U.S. HOUSE OF REPRESENTATIVES
WASHINGTON, D.C. 20515

August 8, 1979

Honorable Joseph Hendrie
Chairman, Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Chairman:

I appreciate your having promptly provided me a copy of the report prepared by the Office of Inspection and Enforcement on the Three Mile Island Accident (NUREG-0600).

Questions have been raised as to the basis for the conclusion of the NRC staff that TMI operators failed to follow procedures when they limited high pressure injection flow. The conclusion that procedures were not followed does not appear to be supported by several NUREG-0600 statements to the effect that operators did in fact act in accordance with their training, and that they did not perceive that there had been a loss of coolant accident. In view of this, I think it important that you provide the Subcommittee with an analysis indicating the basis for the conclusion that the operators should have been using and following the directions in those portions of the operating procedures that pertain to a loss of coolant accident.

I believe also that there should be clarification as to whether procedures were violated by virtue of the operators not having tripped the reactor coolant pumps fifteen minutes after the accident began. NUREG-0600 suggests that there might have been a violation while I&E Bulletin 79-05A implies, to the contrary, that operators erred by prematurely tripping the reactor coolant pumps, which, "... led to fuel damage..." I&E Bulletin 78-05A also directs that operating procedures at B&W facilities be reviewed to insure that, "... in the event of HPI initiation, with reactor coolant pumps (RCP) operating, at least one RCP per loop shall remain operating."

In order that persons having an interest in this matter have full access to the information on which NUREG-0600 is based, I think it important that all reference documents pertaining to it be placed in the Public Document Room.

Sincerely,

MORRIS K. UDALL
Chairman

R.L. TEDESCO, DSS
Phillips
fm: N. C. Moseley, ROI, HQ

DRAFT 9/11/79

9 Pages

URGENT

The Honorable Morris K. Udall, Chairman
Committee on INterior and Insular Affairs
United States Hours of Representatives
Washington, D. C. 20515

Dear Mr. Chairman:

This is in response to your letter dated August 8, 1979, which raised questions regarding the Office of Inspection and Enforcement investigative report of the Three Mile Island Accident (NUREG-0600).

Your first question requested an analysis to support the conclusion in the report that operators should have been following procedures that pertain to a loss of coolant accident. This analysis is provided in Enclosure 1. The statement in NUREG-0600 (Section I 2.15.1) which indicates that operators' actions to limit the high pressure injection flow were influenced by their training was provided as an explanation, but was not intended to imply that this action was in accordance with the licensee's procedures. The investigation findings show that the preponderance of instrument indications of plant parameters should have caused the operator to utilize Procedure 2202-1.3 which is entitled "Loss of Reactor Coolant/Reactor Coolant System Pressure."

Prior statements of the NRC staff, particularly the Lessons Learned Task Force report ~~(Enclosure 3)~~ and testimony to the President's Commission ~~(Enclosure 4)~~, and statements before the ACRS Ad Hoc Subcommittee on TMI-2 Accident Implications, have referred to the inadequacies or ambiguities of the TMI-2 procedures. The conclusion stated in NUREG-0600 about following Procedure 2202-1.3 is not inconsistent with these statements. ~~Nevertheless, it is~~ Nevertheless It is Also not inconsistent with the B&W internal memoranda which recommended that operators be told that pressurizer level could be misleading. This procedure was deficient in that it did not specifically caution the operators that in some circumstances, including a leak from the pressurizer steam space, the pressurizer level may not be a reliable indicator of the primary system inventory. Had this caution been included, the procedure would have been a better one, and would have aided the operators in reaching the proper diagnosis. The collection of emergency procedures may be considered to be inadequate or ambiguous when taken in conjunction with the operators' state of training for emergency actions. These deficiences allowed the operators to be misled by plant parameter indications, the preponderance of which should have ~~signaled~~ led them to conclude that there was a small break accident. Despite ~~these~~ the procedural inadequacies, if the operators had utilized the specific procedure for small break accidents (2202-1.3), it is unlikely that core damage would have resulted.

Emergency procedures for such events as within-containment leaks in the primary or secondary system of necessity contain a spectrum of symptoms. Some of the symptoms are potentially indicative of several occurrences. All of them are not expected to occur during each event. For instance, the core flood tank levels would not decrease unless the reactor coolant pressure reduces to the level where they can inject water. Furthermore, some may be present at one time during an event then go away. One reason operators need intensive training is to enable them to interpret the sympton information presented and make the right decision on corrective action. In this case, the operators rationalized the persistent symptoms and concluded that a loss of coolant had not been experienced. The most important symptom, low reactor coolant pressure, was disregarded in view of the high pressurizer level. Better training, particularly about the idiosyncrasies of very small size loss of coolant accidents, would also have given the operators an improved potential to have understood what was happening. This better understanding could have made them less willing to rationalize the symptoms and ignore the most significant one. This understanding would have counteracted the mind set which gave overwhelming importance to pressurizer level alone.

Notwithstanding the above, as was stated in the Foreword of NUREG-0600, ". . . had certain equipment been designed differently, it could have prevented or reduced the consequences of the accident. The results of the investigation make it difficult to fault only the actions of the operating staff."

The second question requested clarification as to whether procedures were violated by the operators' failure to trip the reactor coolant pumps fifteen minutes after the accident began. As your letter suggests, there exists an apparent conflict between the operators' failure to trip the reactor coolant pumps and the requirements of Bulletins 79-05A and 79-06A, which directed that at least one reactor coolant pump be maintained operating. A subsequent position has been adopted by the NRC and transmitted to licensees on July 26, 1979, via Bulletins 79-05C and 79-06C which supersedes 79-05A and 79-06A? and directs licensees to trip all operating reactor coolant pumps upon receipt of a reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure. The basis for the change in this position is discussed in Enclosure 2. While the operators did not follow the specific requirements of the Emergency Procedure, the failure to shut off the reactor coolant pumps did not of itself cause the accident, but may have contributed to its severity. It should be emphasised, however, that we have not yet decided whether or not to consider this action an item of noncompliance. In all cases in NUREG-0600, potential noncompliance was labeled "under consideration as a potential item of noncompliance." This was done because in an accident situation, unlike during normal operations, one must give consideration to extenuating circumstances such as why actions were taken. The ambiguity which you mentioned in your letter must certainly be a consideration in our final decision on the appropriateness of citing the utility for

Your last question was directed to the availability of the reference documents in the Public Document Room. All but a few miscellaneous documents are expected to be in the Public Document Room by September 17, 1979. The exceptions are cases where proprietary review and clearances from privacy act considerations remain outstanding. We are attempting to clear these few remaining documents as rapidly as possible.

Should you have further equestions on the investigative report of the Three Mile Island Accident, we will be pleased to answer them..

Sincerely,

Joseph M. Hendrie
Chairman

Enclosures:
1. "Limiting High Pressure Injection Flow"
2. "Tripping the Reactor Coolant Pumps"
3. Lessons Learned Task Force Report
4. Testimony to the President's Commission

...iting High Pressure Injection Flow

The operator action to limit HPI flow was not in accordance with TMI Unit 2 Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure" (copy attached). This procedure (in Section B) lists eight symptoms indicative of a leak or rupture of sufficient size such that the Engineered Safety Features Systems, including high pressure safety injection, are automatically initiated. Such an automatic initiation did occur at the beginning of the TMI accident (details in NUREG-0600, Section I2.5). Five of these symptoms existed prior to the time that the reactor fuel became uncovered. These were:

1. Rapid, continuing decrease of reactor coolant pressure.
2. High reactor building ambient temperature.
3. High reactor building sump level.
4. High reactor building pressure.
5. Rapidly decreasing make-up tank level.

The three listed symptoms that did not exist were:

1. Rapid decrease of pressurizer level (after an initial rapid decrease, attributed by the operators to shrinkage from cooldown, the level went up, due to flow out of the pressurizer relief valve, resulting in a misleading indication).

2. High radiation in the reactor building (due to the relatively low level of primary coolant radioactivity at this time).

3. Decreasing core flood tank level and pressure (which would not be expected to occur, as the minimum pressure experienced during the early phases of the accident was 660 psi, and the core flood tanks begin to inject at 600 psig).

The preponderance of evidence available to the operators, therefore, was indicative of a reactor coolant loss.

The Emergency Procedure repeatedly states the necessity of maintaining both pressurizer level and RCS pressure above the 1640 psig safety injection initiation point. Item A3.2.5, for example, specifically cautions that, if the level cannot be maintained above 200 inches and pressure cannot be maintained above 1640 psig, the plant has suffered a major rupture and requires operation in accordance with the section of the Procedure (Part B) applicable to this condition. This section requires establishing an HPI flow of 250 gpm to each of the four reactor coolant legs (125 gpm if one HPI pump fails to start). Contrary to this requirement, although the pressure remained below 1640 after the first 3 minutes of the accident, the net addition rate to the RCS was reduced to an average of about 25 gpm during most of the first 3-1/2 hours. (NUREG-0600 Section 4.3.2.2)

Attached:
TMI Emergency Procedure
2202-1.3

Tripping The Reactor Coolant Pumps

Item B 2.2.4 of TMI Unit 2 Emergency Procedure 2202-1.3, requires that the reactor coolant pumps be tripped before the reactor coolant pressure is reduced to 1200 psi. Since this pressure was reached about 16 minutes after the start of the event and the reactor coolant pumps were not tripped until 101 minutes into the event, this item in the procedure was not followed.

On April 5, 1979 IE Bulletin 79-05A (Attachment 1) was issued to all holders of operating licenses for B&W designed reactors. The Bulletin identified six potential human, design and mechanical failures which resulted in the core damage at TMI-2. These identified failures were based on all of the preliminary information received by the NRC up to that time. Item six on this list stated "Tripping of reactor coolant pumps during the course of the transient, to protect against pump damage due to pump vibration, led to fuel damage since voids in the reactor coolant system prevented natural circulation." Based upon our understanding at that time, one of the actions we required to be taken by licensees was to ensure that operating procedures were revised, if necessary, to specify that in the event of high pressure injection (HPI) initiation, with reactor coolant pumps (RCPs) running, at least one RCP per loop should remain operating. On April 21, 1979, IE Bulletin 79-05B (Attachment 2) was issued to all B&W licensees. This Bulletin provided additional information on the subject of natural circulation. We stated that the preferred mode of core cooling, following a transient or accident, was to provide forced flow using RCPs. It was our opinion that natural circulation was not successfully achieved at TMI-2 upon securing RCPs because of significant coolant voids, possibly aggravated by the release of noncondensible gases in the primary coolant system. Similar requirements to maintain forced flow in a LOCA situation was issued to licensees for reactors designed by other PWR vendors: IE Bulletin 79-06A (April 14, 1979) for Westinghouse and IE Bulletin 79-06B (April 13, 1979) for Combustion Engineering designed plants (Attachments 3 and 4).

Previous analyses had demonstrated that the reactor core would be adequately cooled during a small break LOCA provided the RCPs were tripped at the beginning of the accident. At the request of the NRC, B&W performed additional analyses, for various break sizes, assuming that RCPs were operating at the beginning of the accident and were then inadvertently tripped a short time into the accident. B&W's calculations showed that for a range of break areas between 0.025 and 0.2 ft.^{2}, if RCPs remained operating until the reactor coolant system (RCS) contained a high void fraction and were then tripped, the core would be uncovered for an extended period of time and that under certain conditions, 10 CFR Part 50 ~~Appendix K limits would be exceeded~~. A delayed pump trip can produce higher cladding temperatures than an immediate trip because while the pumps are running, liquid and steam in the primary system are being circulated in a mixed condition. This circulation allows the break to continuously discharge a mixture of water and steam rather than in the immediate pump trip case which eventually discharges only steam. Thus, the liquid inventory in the primary system will be less later in the accident for the pumps running case. If the pumps are either turned off or fail later in the accident, the reduced liquid inventory would result in more extensive and prolonged core uncovery than for the immediate pump trip case. However, liquid separation and fallback could occur even with the pumps running producing degraded core cooling conditions.

Para. 50.46.b (peak cladding temperature may exceed 2200°F)

Staff discussions with vendors on the B&W analysis results concluded that while not all vendors could agree that Appendix K limits might be exceeded in a similar situation for their plant designs, they did agree that if the RCPs were tripped prior to the formation of a high void fraction developing in the RCS, adequate core cooling could be demonstrated. Based upon a review of the analyses presented, that continued operation of the RCPs throughout the entire accident could not be guaranteed, and that adequate core cooling even with the pumps running throughout the accident had not been demonstrated for all conditions, the staff concluded that the proper course of action for an operator to take during a LOCA situation was to trip all operating RCPs as an immediate action.

On July 26, 1979, IE Bulletins 79-05C & 79-06C (Attachment 5) were issued to all PWR licensees. These Bulletins directed the licensees to ensure that upon reactor trip and initiation of HPI caused by low RCS pressure, the operators were instructed to trip all operating RCPs. In addition, the Bulletins required that two licensed operators be present in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurrence. The Bulletins also required additional analyses, development of guidelines and emergency procedures, and operator training in this area to be performed in the short-term (all actions completed by October 31, 1979). As a long-term requirement of the Bulletins, each licensee was directed to submit a proposed design change which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

Attachments:
1. IE Bulletin 79-05A
2. IE Bulletin 79-05B
3. IE Bulletin 79-06A
4. IE Bulletin 79-06B
5. IE Bulletins 79-05C & 79-06C

A. Silver
From SER Input File
TMI 10/30/75

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 30 1975

Docket No. 50-320

V. A. Moore, Assistant Director for Light Water Reactors, Group 2, RL

SAFETY EVALUATION REPORT: THREE MILE ISLAND NUCLEAR STATION, UNIT 2, QUALITY ASSURANCE BRANCH

Plant Name: Three Mile Island Nuclear Station, Unit 2
Licensing Stage: OL
Docket No. 50-320
Responsible Branch: LWR #2-2
Project Manager: B. Washburn
Requested Completion Date: June 27, 1975
Applicant's Response Date: N/A
Description of Response: N/A
Review Status: Complete

The QA Branch has reviewed and evaluated Sections 13.1, 13.4, 13.6, and 17 of the FSAR (through Amendment 28) for Three Mile Island Nuclear Station, Unit 2. Our SER input for Section 13.6 and 17 is enclosed.

We have not included an input for Sections 13.1 and 13.4 for the following reasons:

1. The applicant has not been responsive to our requests for information relative to his offsite technical support for the operation of Three Mile Island, Unit 2. We are therefore unable to reach a conclusion as to the acceptability of this technical support.

2. In Amendment 28, the applicant revised the description of his plant staff. This revision deletes the number of persons assigned to each plant staff position. We are therefore unable to reach a conclusion as to the acceptability of the plant staff.

   The applicant has been advised of these two deficiencies. (Verbally) [illegible] of this 10/30/75

3. The applicant has submitted a proposed revision to the review and audit provisions of Section 6.0 of the technical specifications for Three Mile Island 1. We are reviewing this submittal for conformance to the Regulatory position set forth in Regulatory Guide 1.33 and for consistency with Section 6.0 of the NRC Standard Technical Specifications.

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Upon staff approval of this revision, the applicant will amend the Three Mile Island 2 application to include these review and audit provisions. We consider this to be an acceptable approach.

An SER supplement will be issued when the above matters are resolved.

Donald J. Skovholt
Donald J. Skovholt, Assistant Director
for Quality Assurance & Operations
Division of Reactor Licensing

Enclosure:
As stated

cc: w/o enclosure
W. McDonald

w/enclosure
K. Kniel
B. Washburn

SAFETY EVALUATION REPORT

THREE MILE ISLAND NUCLEAR STATION, UNIT 2

OPERATIONAL QUALITY ASSURANCE PROGRAM

13.0 Conduct of Operations

13.6 Plant Records

The applicant has described his program for maintaining plant records and has committed to maintaining records according to ANSI N45.2.9-1974, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants." Specific records and their retention periods will be delineated in the facility technical specifications.

Based on our review, we conclude that the applicant's provisions for maintaining records meet the position described in ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants," and are satisfactory.

17.2 Quality Assurance For Operations

Organization

Metropolitan Edison Company (MET-ED) has established an organization which is responsible for establishing and implementing the operational QA program for the Three Mile Island Nuclear Station,Unit 2. The President of MET-ED has delegated to the Operational QA Manager, through the Vice President-Generation, the responsibility for establishing and implementing the QA program. As shown in Figure 1, the Operational QA Manager has equal organizational level with the Managers of Engineering, Nuclear Generating Stations, and Maintenance. The onsite Plant QA Supervisor and QA Specialists are under the direct control of the Operational QA Manager.

The qualifications, duties, responsibilities, and authority for the various individual positions performing QA functions have been adequately described and are acceptable. The Operational QA Manager has the specific responsibility to develop, implement, and maintain the operational QA program and manual. QA program procedures are reviewed and approved by the Operational QA Manager. QA related procedures, originated by other MET-ED organizations, are reviewed and approved by the respective organizations and reviewed and concurred in by the Operational QA Manager. To assure continuous implementation of the QA program policies and procedures, the Operational QA Manager conducts a system of preplanned audits, inspections,and review activities. In addition, the Vice President-Generation performs a review and audit evaluation of the QA program effectiveness at least every two years and reports the results to the MET-ED President. We find that the QA organization has adequate authority to identify quality problems; initiate, recommend or provide solutions; and verify implementation of corrective action for nonconforming items or activities. This authority includes the right to stop work.

Based on our evaluation, we conclude that the MET-ED QA organization has the sufficient organizational freedom necessary to effectively execute their QA responsibilities without undue influences of cost and schedule. We have therefore determined that this organizational arrangement is acceptable and complies with the requirements of Appendix B to 10 CFR Part 50.

Quality Assurance Program

MET-ED has committed in the FSAR to structure and implement their QA program in accordance with the NRC guidelines contained in NRC documents WASH 1284, "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants," WASH 1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," and WASH 1283, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," with the exception of certain areas which are described by alternatives which we have evaluated and found acceptable. The QA program provides for a formal training program for those personnel performing QA related activities to assure they are knowledgeable as to the proper interpretation and implementation of the QA manual including its requirements and implementing procedures. In addition, the QA program provides for the necessary controlled procedures which describe how each of the eighteen criteria of Appendix B to 10 CFR Part 50 will be complied with. MET-ED requires a formalized inspection program to be established and implemented by qualified QA personnel independent of the personnel or group performing the work being inspected. This also applies to procurement sources. Provisions are provided to assure inspection instructions describe the method of inspection, the accept and reject criteria, and the degree of documenting and verifying the inspection results.

The audit program provides for regularly scheduled audits of the operation of Three Mile Island Unit 2 and for the prompt reporting of audit results and corrective actions to responsible management levels for their review and assessment. The audit program is under the direction of the Operational QA Department, the Plant Operations Review Committee and the General Office Review Board. To assure proper visibility of problem areas and implementation of corrective action, audit results are distributed to responsible members of management. In addition to the audit program, the Vice President-Generation performs an independent review of the QA program procedures and activities at least once every two years to assure that the QA program is meaningful and effective.

Conclusion

In summary, the staff has determined that MET-ED's QA program for Three Mile Island Nuclear Station, Unit 2, as described in the FSAR through Amendment 28, provides a comprehensive system of planned and systematic controls which adequately demonstrate compliance with each of the eighteen criteria of Appendix B to 10 CFR Part 50. In addition, MET-ED has described an acceptable QA organization which has sufficient authority and independence to permit effective implementation of their QA program without undue influences from costs and schedules. We therefore conclude that the MET-ED QA program is acceptable for control of the quality related activities during the operational phase of the Three Mile Island Nuclear Station, Unit 2.

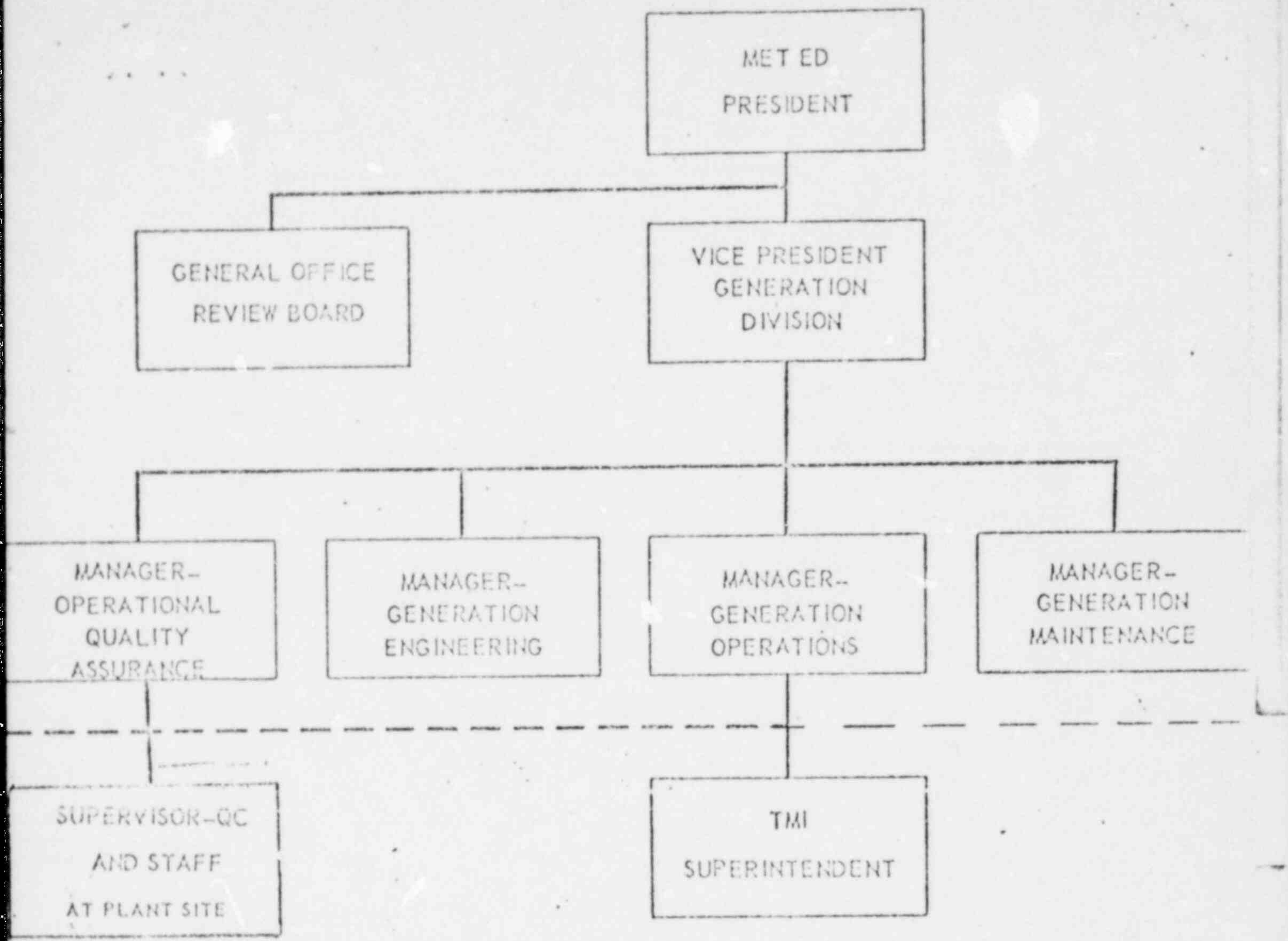

FIGURE 1
ORGANIZATION CHART
THREE MILE ISLAND NUCLEAR STATION UNIT 2

H. VANDERMOLEN

Item 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 1977

Docket No: 50-289

RECEIVED APR 21 1977

MEMORANDUM FOR: K. R. Goller, Assistant Director for Operating Reactors, DOR

FROM: D. G. Eisenhut, Assistant Director for Operational Technology, DOR

SUBJECT: SAFETY EVALUATION - THREE MILE ISLAND UNIT 1 CYCLE 3 RELOAD

PLANT NAME:	Three Mile Island Unit 1
DOCKET NO.:	50-289
RESPONSIBLE BRANCH AND PROJECT MANAGER:	ORB-4 G. Zwetzig
OT BRANCH INVOLVED:	Reactor Safety Branch
DESCRIPTION OF REVIEW:	SER
REQUESTED COMPLETION DATE:	April 5, 1977
REVIEW STATUS:	Complete

The Reactor Safety Branch has reviewed the available information pertaining to the Three Mile Island Unit 1 cycle 3 reload. We have concluded that it is acceptable for the licensee to proceed with the reload in the manner proposed. Our detailed safety evaluation is enclosed.

Darrell G. Eisenhut

Darrell G. Eisenhut, Assistant Director
for Operational Technology
Division of Operating Reactors

Enclosure:
As stated

cc: V. Stello
R. Baer
F. Coffman
C. Berlinger
S. Weiss
M. Chatterton
R. Reid
G. Zwetzig
H. Vander Molen

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Safety Evaluation

License No. DPR-50

Docket Number 50-289

Three Mile Island Nuclear Station, Unit 1

Cycle 3 Reload

Introduction

By letter dated January 26, 1977[1], Metropolitan Edison Company (the licensee) requested changes in the technical specifications appended to Operating License DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1). The proposed changes relate to the discharge of the batch 2 fuel assemblies and replacement with fresh batch 5 assemblies plus assemblies saved from cycle 1a, thus constituting refueling of the reactor for operation in cycle 3. In addition, the proposed changes include operating limits based on an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR Section 50.46.

Reload Description

The TMI-1 reactor core consists of 177 fuel assemblies, each with a 15x15 array of fuel rods. The reload in preparation for cycle 3 operation[2,3] consists of the removal of all batch 2 assemblies, the relocation of batch 3 and 4 assemblies, and the introduction of 13 batch 1a and 48 new batch 5 assemblies. The batch 5 assemblies will be located at the core periphery and the batch 1a assemblies will occupy 13 positions within the mixed central zone.

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Fuel Mechanical Design

The outside dimensions and configuration of the new Mark B-4 (Batch 4 & 5) fuel assemblies and older Mark B-3 (Batch 3) fuel assemblies are identical except that the Mark B-4 have spring-type flexible spacers and the Mark B-3 have corrugated-type flexible spacers. This new fuel rod spacer has been previously reviewed and found acceptable by the NRC staff on the basis of no significant mechanical or material change to the reactor operation[4] and has been successfully operating in similar cores for a substantial time (Reference Section 4.5 of Reference 1). The new Mark B-4 fuel assemblies, therefore, do not represent any unreviewed or untested change in mechanical design from the reference cycle and are therefore acceptable.

This mechanical design change has been taken into account in the various analyses which are discussed in the following sections. The results of these analyses have shown that this fuel design difference in the TMI-1 core is of negligible effect.

Fuel rod cladding creep collapse analyses were performed for the cycle 3 core. The CROV computer code was used to calculate the time to fuel rod cladding creep collapse[1,5]. The calculational methods, assumptions, and data have been previously reviewed and approved by the NRC staff[6]. The analysis assumed a 2000 hour densification time which maximizes creep; no fission gas production which maximizes differential pressure; and a lower tolerance limit on clad thickness and an upper tolerance limit on cladding ovality, both of which maximize cladding creep deformation.

The batch 3 fuel was found to be more limiting than the batch 4, 5, and 1a fuel due to the lower prepressurization, lower pellet density, and previous power history. The most limiting assembly in batch 3 was found to have a collapse time longer than the maximum projected three-cycle core exposure (24,288 EFPH).

From the viewpoint of cladding stress due to differential pressure, thermal stress due to fuel temperature gradients, and bending stress, neither the yield stress nor the B&W 1% total strain criterion for the cladding is predicted to be exceeded in the cycle 3 core.

The Batch 5 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previously successful operations with equivalent fuel the staff concludes that the fuel mechanical design for cycle 3 operation is acceptable and its application to cycle 3 operation will not endanger the health and safety of the public.

Fuel Thermal Design

The fuel thermal design analysis was conducted with the TAFY-3 computer code, as discussed in reference 7. The analysis considered the effect of a power spike from fuel pellet densification, as modeled in the "Fuel Densification Report"[8]. Modifications to the "Fuel Densification Report" on the fuel pellet void probability, F_g, and fuel grain size distribution, F_k have been previously reviewed and approved by the NRC staff.[9]

Based on the analyses presented in reference 1 and comparison with allowable Linear Heat Generation Rate (LHGR) for fuel centerline melt considerations, the fuel thermal design for the cycle 3 core is acceptable and can be applied with reasonable assurance that the health and safety of the public will not be endangered.

Fuel Material Design

The fuel material design for cycle 3 operation is not significantly different from that of cycle 2 operation. The only difference is that Zircaloy-4 is used as the fuel assembly tubular spacer material in Mark B-4 fuel instead of zirconium dioxide (ZrO_2), which is used in Mark B-3 fuel. This change does not affect the fuel system chemistry. This change has been reviewed and has a substantial amount of previous experience (Section 4.5 of reference 1). Therefore, the fuel material design for TMI cycle 3 operation is acceptable.

Nuclear Design

The TMI-1 reactor has completed two operating cycles and is thus sufficiently close to equilibrium cycle to show only minor changes in physics parameters. The cycle 3 core will consist of four distinct fuel types: fresh batch 5 assemblies located at periphery, once-burned batch 4 assemblies located generally in an intermediate zone and also near the core center, twice-burned batch 3 assemblies located between the periphery and the intermediate zone, and located between the intermediate zone and central zone, plus 13 batch 1a assemblies loaded with the batch 4 assemblies. Thus, although the cycle 3 core is a four batch loading, the physics parameters are quite close to those of the cycle 2 core. In addition, these parameters will be verified during the startup testing program described later.

The only significant procedural change from the reference cycle (cycle 2) is the specification of axial power shaping rod (APSR) position limits. The APSR position limits will provide additional control of power peaking through an improved definition of the core power distribution.

The calculational methods used by the licensee are the same as were used for cycle 2.[10] Because of this, and because of the verification provided by the physics testing which will be performed during the cycle 3 startup, the staff finds the nuclear design for cycle 3 to be acceptable.

Thermal-Hydraulic Analysis

Major acceptance criteria for the thermal-hydraulic design are specified in the NRC's Standard Review Plan Section 4.4 ("Thermal and Hydraulic Design"). These criteria establish the acceptable limits for DNBR (Departure from Nucleate Boiling Ratio). The thermal-hydraulic analyses for the TMI-1 cycle 3 reload core were made with previously approved models and methods, as stated in the TMI-1 Final Safety Analysis Report[11].

The reactor coolant flow rate was accurately measured during cycle 1 operation and a minimum measured value of 108% of the system design flow was determined. The licensee has taken credit in the cycle 2 & 3 thermal-hydraulic analyses for the fact that the actual system flow is greater than the design flow rate, and has also included uncertainties and conservatisms in this analysis.[1, 10] The new design flow is 106.5% of the cycle 1 design flow.

In the past, a reactor coolant flow penalty had been assumed in the thermal-hydraulic design analysis for TMI-1. This penalty was associated with the potential for a core internal vent valve to be stuck open during normal operation. The core internal vent valves are incorporated into the design of the reactor internals to preclude potential vapor lock during a postulated cold-leg break Loss-of-Coolant Accident (LOCA). The NRC staff has concluded that by application of a surveillance program the vent valve flow penalty may be removed. The surveillance requirements demonstrate that the vent valves are not stuck open and that the vent valves operate freely. A separate review of the Licensee's surveillance program for the vent valves has concluded that the program adequately meets the staff's requirements, and that the vent valve penalty was properly eliminated[12].

The effect of fuel rod bow was evaluated by the Licensee with consideration given to both the hot channel power spike and the effect on DNBR. This evaluation was also separately reviewed and accepted by the staff[12].

There are differences in the flow resistance between the Mark B-3 fuel assemblies and the Mark B-4 assemblies. The flow resistance for a Mark B-4 fuel assembly is slightly less than that for the Mark B-3 assemblies. For the cycle 3 loading, the highest assembly power always occurs in a Mark B-4 assembly. The cycle 2 analysis[10], also used for cycle 3 reference evaluation[1], assumed the hot assembly to be a Mark B-3 type. This analysis is conservative for cycle 3 because the predicted hot assembly coolant flow rate is less than that of a corresponding Mark B-4 assembly.

Because of the analyses discussed above, we have found the thermal-hydraulic analysis to be acceptable and the proposed Technical Specifications related to the thermal-hydraulic analysis also acceptable.

Accident and Transient Analyses

A generic LOCA analysis for a B&W 177 assembly lowered-loop plant has been performed using the Final Acceptance Criteria ECCS evaluation model[13, 3]. This analysis has been reviewed by the staff[14], and found applicable to the TMI-1 cycle 3 core.

All other accidents and transients (loss of flow, dropped rod, inadvertent bank withdrawal, etc.) have been examined by the licensee for cycle 3 and found to fall within the bounds of the FSAR analyses, as updated for cycle 2 operation. The staff has reviewed the various input parameters for cycle 3, and has found the licensee's conclusion acceptable.

Startup Program

The licensee has proposed a startup program which will verify:

. Critical boron concentration

. Temperature reactivity coefficient at two points

. Control bank worth by boron swap. More than half of the required shutdown reactivity will be verified

. Control bank worth by bank drop. The remainder of the banks will be checked by this method.

. Ejected rod worth

In addition, during the power escalation phase, the startup program will verify:

. Power distribution at three plateaus.

. Dropped-rod power distribution

. Incore/excore imbalance correlation

. Doppler coefficient at 100% power

. Temperature reactivity coefficient at 100% power

The staff has reviewed this proposed startup program and has found it acceptable.

Technical Specifications

The licensee has proposed revisions to the technical specifications to implement the changes due to the cycle 3 reload[1]. The staff has reviewed the revised technical specifications and found them acceptable except for the following modifications, which we will require and to which the licensee has agreed:[3]

. Add the following:

3.1.7.2 The moderator temperature coefficient shall be $\leq + 0.5 \times 10^{-4}$ $\Delta k/k/F$ at power levels $\leq$ 95% of rated power.

. Revise 3.5.2.7 to read:

3.5.2.7 A power map shall be taken at intervals not to exceed 30 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in Figure 3.5-2J.

Conclusion

Based on our evaluation of the application and available reload information as set forth above, and assuming compliance with the requirements set forth above, we conclude that it is acceptable for the licensee to proceed with cycle 3 operation in the manner proposed.

References

1. Letter, R. C. Arnold (Metropolitan Edison) to Director of Nuclear Reactor Regulation, dated January 26, 1977, enclosing Technical Specification Change Request No. 45 and BAW-1442.

2. BAW-1442, Three Mile Island Unit 1 Cycle 3 Reload Report, November, 1976.

3. Letter, R. C. Arnold (Metropolitan Edison) to Director of Nuclear Reactor Regulation, dated March 31, 1977, enclosing responses to Round 1 Questions concerning the TMI-1 cycle 3 Reload Application.

4. SER on Oconee Nuclear Station, Units 1,2,&3, dated June 30, 1976, Amendment Nos. 27, 27, and 23 for License Nos. DPR-38, DPR-47, and DPR-55.

5. BAW-10084P, Rev. 1, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, October, 1976.

6. Letter, A. Schwencer (NRC) to J. F. Mallay (B&W), dated January 29, 1975.

7. BAW-10044, TAFY - Fuel Pin Temperature and Glass Pressure Analysis, May, 1972.

8. BAW-10055, Rev. 1, Fuel Densification Report, June, 1973.

9. Memorandum from R. Lobel to D. F. Ross, "Present Status of B&W Power Spike Model," July 23, 1974.

10. Three Mile Island Unit 1 - Cycle 2 Reload Report, Rev. 1, July, 1976.

11. Three Mile Island Unit 1 Nuclear Station, Final Safety Analysis Report, USNRC Docket No. 50-289.

12. Safety Evaluation by the Office of Nuclear Reactor Regulation, Amendment No. 25 to Facility Operating License No. DPR-50, dated March 7, 1977.

13. BAW-10103, Rev. 1, ECCS Analysis of B&W's 177-FA Lowered Loop NSS, September, 1975.

14. Letter, D. F. Ross (NRC) to D. B. Vassallo (B&W), Re: Topical Report Evaluation BAW-10104, ECCS Evaluation Model, Revised Nucleate Boiling Lockout Model, dated February 2, 1977.