Safety Evaluation Report

NUREG-0107
SUPP. 2

U. S. Nuclear Regulatory Commission

related to operation of

Three Mile Island Nuclear Station, Unit 2

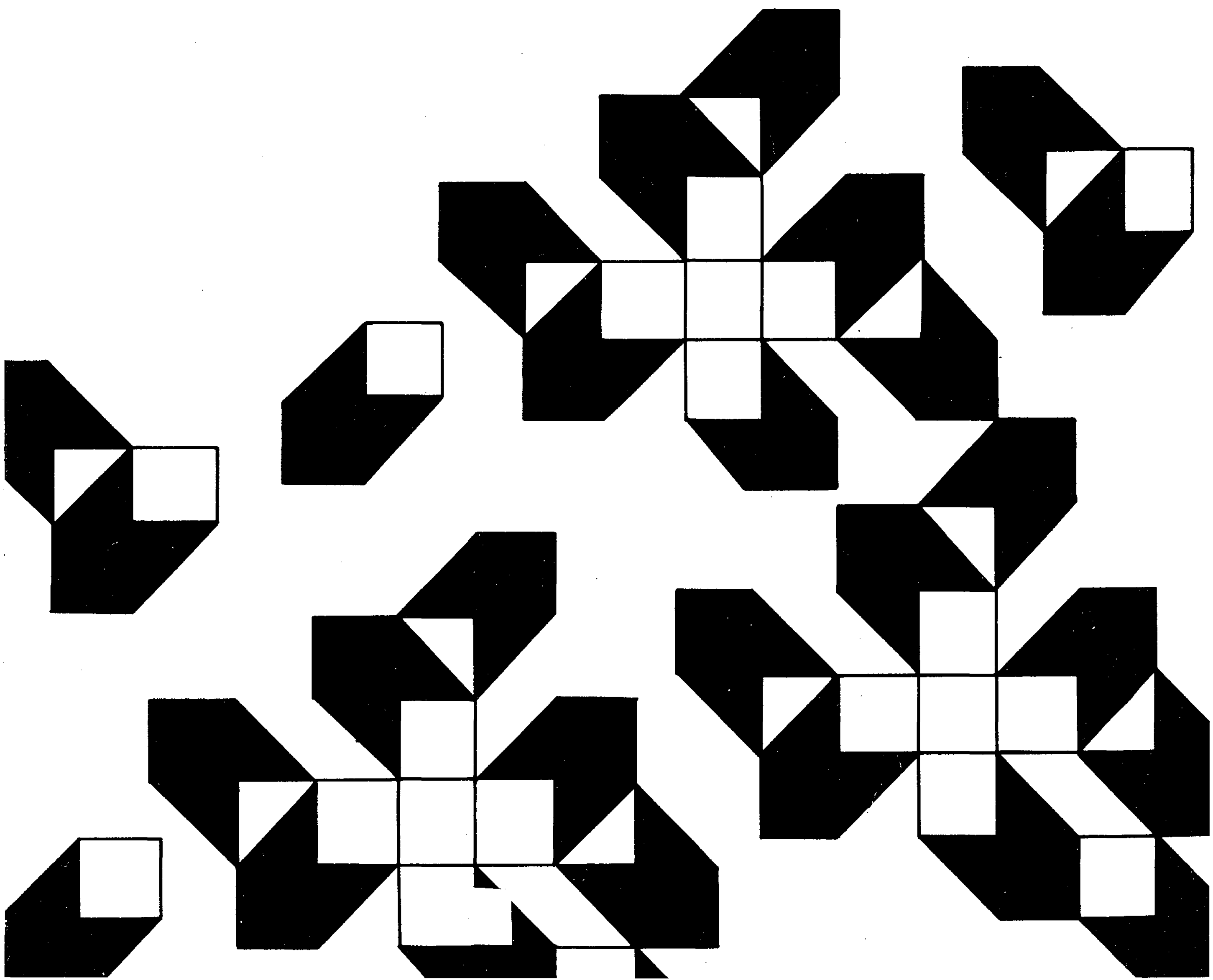
Office of Nuclear Reactor Regulation

Docket No. 50-320

Metropolitan Edison Company
Jersey Central Power & Light Company
Pennsylvania Electric Company

February 1978

Supplement No. 2



Available from
National Technical Information Service
Springfield, Virginia 22161
Price: Printed Copy $5.25; Microfiche $3.00

The price of this document for requesters outside of the North American Continent can be obtained from the National Technical Information Service.

FEBRUARY 1978

SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

UNITED STATES NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION

UNIT NO. 2

DOCKET NUMBER 50-320

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

APPENDICES

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On September 10, 1976 the Nuclear Regulatory Commission (Commission) issued its Safety Evaluation Report regarding the application by Metropolitan Edison Comapny, Jersey Central Power and Light Company, and Pennsylvania Electric Company for licenses to operate Three Mile Island Nuclear Station, Unit 2 (Three Mile Island Unit 2, the plant, or the facility). Supplement No. 1 was issued on March 11, 1977. That supplement documented the resolution of many outstanding items and summarized the status of the remaining outstanding issues.

The purpose of this supplement is to update our Safety Evaluation Report and Supplement No. 1 by providing (1) our evaluation of additional information submitted by the applicant since the issuance of Supplement No. 1 to the Safety Evaluation Report, and (2) our evaluation of additional information for those sections of the Safety Evaluation Report where further discussion or changes are in order.

Each section of this supplement is numbered the same as the section of the Safety Evaluation Report, and is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and the supplements thereto, except where specifically so noted.

In the Safety Evaluation Report, we stated that the present ownership arrangement was defined in Amendment 13 submitted February 14, 1974, and further stated that Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company own Three Mile Island Unit 2 on a 50/25/25 percent basis, respectively. In Amendment 55, submitted April 4, 1977, we were informed that the ownership was changed to 25/65/10 percent, respectively. (See Section 20.0 herein for additional discussion).

18. Restrictions on opening watertight doors between the turbine and control building area (Section 10.4)

1.7 Summary of Outstanding Review Items

All of the items previously identified as outstanding in Supplement No. 1 to the Safety Evaluation Report have been resolved, as noted below. New issues addressed since Supplement No. 1 are so identified. Resolution of some items require limitations on plant operation and are identified accordingly. For all items, additional discussion is presented in the referenced sections of this supplement.

Previously Identified Items - Resolved

-- Reactor cavity pressure response analysis confirmation (Section 6.2.1).

-- Reactor Building Sump Test Results (Section 6.3.4).

-- Installation of flow measurement devices to assure adequate flow to limit boron concentration (Section 6.3.3).

-- Qualification as isolation devices of output relays in the safety features actuation system (Section 8.3.1).

-- Description of test programs and methods for:

    a. Reactor protection system and engineered safety features actuation system response time testing (Section 14.0).

    b. Control rod drop time testing to full insertion (Section 14.0).

-- Emergency core cooling analysis modifications (Section 6.3.3).

Previously Identified Items - License Conditions Required

-- Completion of revised steam line break analysis (Sections 10.3, 10.5, 7.3.3, 7.4.1, 6.2.1, 15.2.2, 15.3.1, 15.3.2).

-- Evaluation of data on environmental qualification of balance-of-plant Class IE electrical equipment (Section 7.8.2).

-- Review of feed line break analysis (Section 15.2.2).

-- Startup overpressure protection (Section 5.2.2).

New Issues - Resolved

-- Irradiation of demonstration burnable poison rods (Section 4.5).

-- Steam generator tube integrity (Section 5.5.2).

-- Reactor cavity cooling fan classification (Section 6.2.2).

-- Fuel handling accident inside containment (Section 15.3.4).

New Issues - License Conditions Required

-- Reactor building emergency cooling booster pump flow (Section 6.2.2).

-- Makeup tank valve closure (Section 6.3.2).

-- Offsite power system degraded grid conditions (Section 8.2).

-- Fire protection system improvements (Section 9.5).

-- Diesel generator start capability (Section 9.6.2).

2.0 SITE CHARACTERISTICS

2.3 Meteorology

2.3.4 Short-Term (Accident) Diffusion Estimates

In the Safety Evaluation Report, we noted that to accommodate uncertainties from less than 80 percent meteorological data recovery, a 20 percent uncertainty factor was incorporated in our dispersion estimates.

The applicant subsequently provided data covering a one-year period (July 1976 through June 1977) with a data recovery rate of 93 percent. Based on these data we reevaluated short-term releases, and found the maximum relative concentration (X/Q) for the 0-2 hour time period following an accidental release to be 8.3×10^{-4} seconds per cubic meter, occurring in the east sector at a distance of 610 meters. We calculated the relative concentrations at the outer boundary of the low population zone (3218 meters) for longer time periods to be:

Time Period	X/Q, Seconds Per Cubic Meter
0-8 hours	6.9×10^{-5}
8-24 hours	4.8×10^{-5}
1-4 days	2.3×10^{-5}
4-30 days	7.5×10^{-6}

2.3.6 Conclusions

In the Safety Evaluation Report, we stated that we required the applicant to submit a year of onsite meteorological data with data recovery of at least 90 percent, and that we would use the data to modify the relative concentration values as appropriate.

As indicated in Section 2.3.4 above, the applicant has submitted the required data, and we have modified the relative concentration values appropriately.

For all time periods other than the 4-30 day period, the relative concentrations above are lower than those in the Safety Evaluation Report. The increase in the 4-30 day relative concentration will cause only an insignificant increase in the loss-of-coolant accident dose.

4.0 REACTOR

4.5 Irradiation of Demonstration Burnable Poison Rods

Since publication of the Safety Evaluation Report and Supplement No. 1, the applicant has proposed to irradiate rods containing two burnable poison materials during Cycle 1 operation at Three Mile Island Unit 2. These burnable poisons, while new to Babcock & Wilcox pressurized water reactors, have seen extensive use in other thermal reactor designs. Boronated graphite burnable poison rods are used in high temperature gas cooled reactors of General Atomic design, such as Fort St. Vrain. Gadolinia - uranium dioxide fuel rods have been used in boiling water reactors for several years and have also seen limited use in other pressurized water reactors as well.

Eight Zircaloy-clad burnable poison rods using boron carbide particles in graphite matrix (boronated graphite) pellets will be employed in place of eight standard burnable poison rods. Approximately 97 percent of the pellet is matrix material. Comparison with burnable poison rods in the reactor shows the dimensions to be identical.

Four fuel assemblies, each containing four gadolinia-uranium dioxide fuel rods (total of 16) are also to be irradiated in Cycle 1. Enrichment in the gadolinia-bearing rods was reduced from 1.98 percent to 1.80 percent (by weight) U-235, to compensate for a reduction in pellet thermal conductivity relative to pure uranium dioxide. The pellet thermal dimensions, total column length, stack weight, cladding material and dimensions, grids and end fittings are identical to those in non-gadolinia bearing fuel assemblies.

Design Evaluation - Boronated Graphite Rods

The principal performance considerations of concern for boronated graphite burnable poison rods involve radiation-induced swelling, gas release, and compatibility with cladding and coolant.

The applicant has shown that these factors have been taken into account in an acceptable manner. We therefore conclude that there is reasonable assurance that cladding integrity will be preserved throughout Cycle 1 operation. However, even if all the boron carbide in the burnable poison rods were rapidly removed via primary coolant ingress through cladding perforations, the neutronic effect would be unimportant because the number of such rods is small and they are well spaced in one spider.

Design Evaluation - Gadolinia-Uranium Dioxide Fuel Rods

The principal performance concerns for gadolinia-uranium dioxide fuel rods involve the effects of the gadolinia additions on material properties such as thermal conductivity and irradiation-induced densification. There is evidence that gadolinia-bearing rods will densify more in-reactor than uranium dioxide rods. Their lower thermal conductivity will also tend to increase pellet temperatures. Nevertheless, taking into account the lower enrichment of the demonstration rods and their location in the core, the high power uranium dioxide rods are probably more limiting.

In summation, because the demonstration rods are relatively few in number and are not located in peak power assemblies, we conclude that no significant safety concern exists regarding the proposed irradiation of gadolinia-uranium dioxide demonstration fuel rods and that there is reasonable assurance that cladding integrity will be maintained throughout fuel Cycle 1 operation.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

In the Safety Evaluation Report, we mentioned reported incidents of reactor vessel overpressurization and indicated we would continue our review of this matter.

For Three Mile Island Unit 2, the applicant has developed administrative procedures and provided equipment to minimize the potential for excessive pressure transients under startup and shutdown conditions. Either a steam or nitrogen bubble will be maintained in the pressurizer with a high level alarm and a low level interlock to maintain specified level limits. The presence of a bubble reduces the repressurization rate which results in more time for operator action. A single dual range relief valve will also be available during the first fuel cycle.

We have performed an evaluation of the Three Mile Island Unit 2 pressure vessel and determined that because of the minimal effects of radiation during the first fuel cycle, the allowable stress limits will not be reduced below stresses resulting from overpressure events limited by safety valve set points with the vessel at ambient temperature. This evaluation provides the principal basis for concluding that an overpressurization event during the first fuel cycle would not exceed the limitations of 10 CFR Part 50, Appendix G, and that existing equipment and procedures regarding overpressurization protection are acceptable for the first fuel cycle.

We have reviewed the dual set point design and the results of the analyses performed by the applicant to determine if adequate protection is provided through the life of the plant. The present design does not meet the single failure criterion because only a single relief valve has been provided. Also, the computer code DYSID used in the analyses has not been reviewed by the staff.

Prior to startup following the first regularly scheduled refueling outage, we require implementation of modifications meeting the following criteria, with appropriate approved analyses. We will condition the operating license to assure such implementation.

(1) Credit for operator action. No credit can be taken for operator action until ten minutes after the operator is made aware that a transient is in progress.

(2) Single failure criteria. The pressure protection system should be designed to protect the vessel, given any event initiating a pressure transient.

Redundant or diverse pressure protection systems will be considered as meeting the single failure criteria.

(3) Testability. Provisions for periodic testing of the overpressure protection system(s) and components shall be provided. The program of tests and frequency or schedule thereof will be selected to assure functional capability when required.

(4) Seismic design and IEEE 279 criteria. Ideally, the pressure protection system(s) should meet both seismic Category I and IEEE 279 criteria. The basic objective, however, is that the system(s) should not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

(5) Reliability. The system(s) provided must not reduce the reliability of the emergency core cooling system or residual heat removal systems.

5.5 Component and Subsystem Design

5.5.2 Steam Generator Tube Integrity

In the Safety Evaluation Report, we concluded that all reasonable measures have been taken to ensure that the steam generator tubing will not be subject to conditions that will cause deleterious wastage and cracking. This was based in part on the evidence at the time that there had been no instances of tube degradation in once-through steam generators such as those employed at Three Mile Island Unit 2.

Since that time, there have been instances of tube damage in such steam generators at other operating plants. Based on preliminary assessments, it appears that this damage may be the result of flow-induced vibration, rather than wastage or "denting" as have been experienced in steam generators of other designs. We consider this problem generic in nature, and have initiated a review of this matter to identify more positively the cause of the problem and possible solutions. Babcock & Wilcox and several utilities using their steam generators have begun programs of analysis, shop tests, and tests in operating steam generators to accomplish these goals.

The applicant has elected to participate in this effort through three investigation and test programs at Three Mile Island Unit 2, as follows:

- A pre-operational baseline eddy current examination of all steam generator tubes

- A steam generator instrumentation program

- A tube sleeving test program.

Baseline eddy current tube examination program

In addition to the examination requirements of the ASME Code, the applicant has performed a pre-operational eddy current examination of all steam generator tubes at Three Mile Island Unit 2.

According to verbal information from the applicant, the eddy current examination revealed that approximately 30 tubes had areas with reductions of wall thickness in excess of 40 percent, and a number of tubes had lesser indications. Four tubes with different kinds of indications were removed and examined in the laboratory by the applicant and Babcock & Wilcox. Based on this examination, it appears that all the indications were manufacturing defects. Thirty-five tubes, including all those with indications exceeding 40 percent, were plugged. This investigation is continuing on a generic basis, with primary emphasis on relating the defects to manufacturing causes. Upon completion of this investigation, appropriate documentation will be provided.

This program involves no physical change to the steam generator, except the plugging of tubes in accordance with our requirements. Defects appear to be a result of manufacturing techniques rather than any environmental conditions. Based on the above information, we conclude that performance of the baseline eddy current examination of the steam generator tubes is acceptable, and that operation with the installed steam generators is acceptable.

Instrumentation program

This program involves the installation in the steam generator and elsewhere in the plant of accelerometers, temperature and flow measurement devices, and readout equipment to provide tube response, flow, and plant process data. These data will be used to evaluate the effects of normal and transient plant operation on the response of tubes and to relate that response to the most plausible steam flow excitation mechanism.

The applicant has submitted descriptions of the installations, stress analyses of the devices internal to the steam generator and the penetrations of the steam generator, and an evaluation of the effects of the test equipment on the safety of the plant.

Based on a review of the information supplied by the applicant and the similarity of this program with those in progress at the Oconee plant, we conclude that neither the structural integrity nor the operational characteristics of the safety-related equipment involved will be affected by the program modifications, nor will the consequences of an accident be increased as a result thereof, and therefore that these modifications are acceptable.

Tube sleeving test program

Both Babcock & Wilcox and Combustion Engineering have designed test sleeves for insertion in the upper end of a number of steam generator tubes, with appropriate instrumentation, to assess their effect in stiffening the tubes to improve their capability to resist flow-induced vibration, the most likely cause of tube damage found in some other Babcock & Wilcox steam generators.

Safety evaluations have been submitted by the applicant for both designs, including descriptions of the sleeves, locations in the steam generator, method of installation, instrumentation, bench test results, stress analyses, and an evaluation of the effects of the test equipment on the safety of the plant.

Based on a review of the information supplied by the applicant and the similarity of the sleeves to installations in some Combustion Engineering steam generators, we conclude that neither the structural integrity nor the operational characteristics of safety-related equipment involved will be affected by the test program modifications, nor will the consequences of an accident be increased as a result thereof, and therefore that these modifications are acceptable.

After completion of all the above programs we will determine what modifications, if any, we will require to be made to Three Mile Island Unit 2.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In connection with the reactor cavity pressure response to a postulated loss-of-coolant accident, we noted in the Safety Evaluation Report that the applicant was redesigning the shield arrangement to include a shadow shield above the reactor, and that the applicant had stated that the detailed design would be such that the reactor cavity pressure response would be bounded by the analysis already performed for the prior design.

Subsequently, the applicant elected to provide a neutron shield consisting of hinged water-filled stainless steel tanks designed to provide adequate reactor cavity pressure relief by swinging aside in the event of a pipe break. The applicant performed a pressure analysis for a postulated double-ended rupture in the reactor coolant cold leg piping. This break has been adequately justified as the most severe for the reactor cavity subcompartment analysis. Based on our review of the information submitted by the applicant and our confirmatory analysis, we conclude that the analysis of the reactor cavity pressure response and design differential pressure for the reactor cavity walls has been performed conservatively and is acceptable. The applicant states that based on the calculated peak pressure in the break node, the ultimate shear capacity of the reactor cavity wall has a factor of safety of approximately three. We therefore conclude that the design of the reactor cavity wall is acceptable, and we consider this matter resolved.

We further noted in the Safety Evaluation Report that we had requested additional information and analyses regarding the containment temperature and pressure response following a postulated main steam line break accident.

The applicant has recently submitted analyses showing a temperature profile with a peak containment temperature of 389 degrees Fahrenheit. The peak containment pressure is well below the design pressure of the containment (60 pounds per square inch gage). Also provided were analyses demonstrating the capability of certain equipment within containment to withstand the steam line break environment. Although we have not yet completed our review of this material, our preliminary assessment indicates that additional information may be required. We will condition the operating license to require acceptable analyses prior to initial criticality.

Based on the above, we conclude that operation prior to initial criticality represents no undue risk to the health and safety of the public, and is acceptable.

6.2.2 Conainment Heat Removal Systems

Containment Spray Pump Net Positive Suction Head

In the Safety Evaluation Report, we noted that the applicant had provided analyses showing that adequate net positive suction head is available for the containment spray pumps and that verification of reactor building sump performance was expected.

As reported in Section 6.3.4 herein, vortex suppression, performance of the sump, and pressure drop in pump suction lines have been shown to be acceptable. However, with regard to the containment spray pumps, additional concerns have arisen since publication of the Safety Evaluation Report and Supplement No. 1.

After failure of the containment spray pumps to meet the vendor's head curve during system tests, the impellers were returned to the vendor and new head curves were generated. These curves satisfy system flow requirements, but during system runout conditions in the recirculation mode the pumps now require about three feet greater net positive sucton head than that available.

The applicant proposes to resolve this situation by throttling flow to design levels in the injection mode and again in the recirculation mode, thus decreasing net positive suction head required, and states there is ample time available after a postulated accident to accomplish these manual actions. System data have been provided to describe the problem, to support the proposed manual action, and to justify pump integrity for short term operation at reduced net positive suction head. This information was received quite recently and we have not yet completed our review. Our preliminary review, however, indicates that although there may be additional information required, we expect satisfactory resolution of our concerns. We will condition the operating license to assure submittal and acceptance of all required information prior to initial criticality.

Based on the above and subject to satisfactory resolution of our concerns, we conclude that adequate net positive suction head for the containment spray pumps will be assured.

Reactor Cavity Cooling Fans

Subsequent to issuance of Supplement No. 1 to the Safety Evaluation Report, the applicant informed us that the two fully redundant reactor cavity cooling fans are not classified as seismic Category I, nor are they served by emergency power sources. These fans cool the reactor vessel foundation and the primary shield wall to maintain the normal temperature of the concrete below 150 degrees Fahrenheit. The applicant states that one fan will accomplish this purpose.

Although we do not require that these fans be safety grade, we expressed concern that failure of both these fans could possibly affect the safety of the plant by permitting concrete temperatures to exceed 150 degrees Fahrenheit. The applicant has indicated that this temperature is conservatively selected, that any effect of exceeding this temperature would be a long-term one, and that this effect would be minimal due to the self-insulating properties of the very thick conservatively designed concrete structures involved. In addition, the applicant states that preliminary results of an analysis they are performing indicate that the peak temperature reached by the concrete without fan cooling and after reactor shutdown as required by the Technical Specifications, is well below that allowed by Section II of the ASME Code, Paragraph CC-3440, for short term or accident conditions.

The applicant proposed and we will incorporate a Technical Specification to assure that at least one cooling fan will be in operation during reactor operating modes, and to define action if at least one fan is not in operation.

Based on the above, we conclude that the present reactor cavity cooling fans are acceptable.

Reactor Building Emergency Cooling Booster Pumps

The applicant also informed us recently that the impellers for the four reactor building emergency cooling booster pumps provide a pump capacity of 2700 gallons per minute versus the specified 3000 gallons per minute. These pumps supply cooling water to the reactor building fan coolers under accident conditions.

Presently required minimum safeguards include one pump serving two fan coolers. Since there are a total of five coolers available to provide two trains of two fan coolers each, the applicant has verbally proposed as an interim measure, pending installation of the correct impellers, that one fan cooler be maintained as an installed spare with the water valves to this unit closed and locked out. Single failure capability is still maintained. The flow provided by the pumps with smaller impellers is adequate for this arrangement. We will condition the operating license to require the applicant to provide acceptable documentation including his proposed solution prior to initial criticality, and to install the correct impellers and return the system to its design configuration within six months following issuance of the operating license.

Based on the capability of the system to satisfy the design criteria during that interim period, and subject to receipt of acceptable documentation, we conclude that operation for a six month period as described above is acceptable.

6.3 Emergency Core Cooling System

6.3.2 System Design

During the course of our review, we became aware of a discrepancy between the Final Safety Analysis Report, which showed the makeup tank isolation valve MU-V-12 actuated by an engineered safety features signal, and the applicant's operating procedures, which required manual closing of the same valve. The applicant informed us that the former was in error; however, this opened the possibility that, in the event of an accident and the failure to close MU-V-12 in a timely manner, the high pressure injection pumps, taking suction both on the makeup tank and the borated water storage tank in the injection mode, could empty the makeup tank and draw its hydrogen cover gas into the pump and the reactor coolant system.

The applicant has committed to automate MU-V-12 and to add a redundant automated valve to preclude single failure defeat of the closing function prior to startup following the first regularly scheduled refueling outage. The closing time of these valves will be compatible with the opening time of the borated water storage tank discharge valve to assure that flow is established prior to isolation of the makeup tank. Automated isolation of the pump recirculation lines will also be provided. We find these changes acceptable, and will condition the operating license to assure implementation prior to startup following the first regularly scheduled refueling outage.

The applicant proposed the following interim arrangement. The high pressure injection pumps will be aligned initially so that one pump will serve as a spare, and that no more than two of the pumps will, under accident conditions, take suction from the borated water storage tank and the makeup tank. This preserves single failure capability while precluding excessive drawdown of the makeup tank. In addition, the hydrogen charging line isolation valves will be made redundant and actuated by an engineered safety features signal to preclude continued hydrogen pressurization in the event of an accident. (For the final modification discussed above, the signal to the second hydrogen valve will be rerouted to MU-V-12.) Makeup tank charging procedures will be controlled to assure that conditions at the time of switchover to the recirculation mode are such that a water seal will still be maintained in the makeup tank discharge piping even if MU-V-12 is not closed. However, existing operating procedures require closing of this valve. These features provide reasonable assurance that in the event of an accident, no damage to the high pressure injection pumps will occur due to makeup tank drawdown, nor will hydrogen gas be injected into the reactor coolant system, and we therefore conclude that these interim changes are acceptable for the first fuel cycle. We will appropriately condition the operating license to require the above modifications for the first fuel cycle.

6.3.3 Performance Evaluation

General

In the Safety Evaluation Report, we noted that all issues identified in our evaluation of Babcock & Wilcox Topical Report BAW-10103, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," including the small and transition break analysis, had not yet been satisfied.

The requested small and transition break analysis has since been provided by Babcock & Wilcox and the applicant has incorporated this documentation by reference. We have completed our review of this material and find that studies of the spectrum of breaks have been completed and are in accordance with the emergency core cooling system acceptance criteria, and are acceptable. We therefore consider this matter resolved.

Long Term Boron Concentration

In the Safety Evaluation Report, we found that the procedures and systems for preventing excessive boric acid buildup in the reactor vessel following a postulated loss-of-coolant accident were acceptable, subject to:

(1) Our review of the design modifications necessary to provide flow measurement capability, and

(2) Demonstration of the required flow rate in each mode.

The applicant has since submitted information describing the flow measurement system, including permanently installed ultrasonic transducers, connnections for remote readout, and portable readout devices. We conclude that the design of this installation is acceptable.

The applicant has committed to perform a demonstration of the required flow rate in each mode during preoperational or startup testing and to verify the calibration of the flow measurement systems. We will assure that this is accomplished prior to initial criticality. We consider this matter resolved.

6.3.4 Tests and Inspections

In the Safety Evaluation Report, we stated we would consider a program of analysis, plant testing, and scale model testing to demonstrate containment sump vortex control, adequacy of available net positive suction head for the emergency core cooling and containment spray pumps, and system cleanliness and valve operability to satisfy the requirements of Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors."

The applicant has completed the model test program and associated analyses, and has adequately documented this information. The one-third scale model of the containment floor and sump demonstrated that the original layout and design might cause formation of potentially damaging vortices. Design modifications involving various vortex suppressors were tested in the model and the selected configuration was shown to provide adequate vortex control. The testing laboratory and the applicant concluded that the results may safely be projected to the full scale facility, and we concur. The Final Safety Analysis Report has been revised to show these modifications. We consider the matter of sump vortex control to be resolved.

We accept the applicant's stated commitment to verify system cleanliness and valve operability during preoperational testing, and consider these matters resolved.

With regard to adequacy of net positive suction head at the pumps, the applicant has satisfactorily combined entrance loss data from the sump model test with measured pressure drop data in that part of the pump suction lines which could be measured during preoperational tests to show that the calculated pressure drops in the pump suction lines are conservative. We consider this matter resolved. Additional discussion of the containment spray pump net positive suction head appears in Section 6.2.2 herein. We conclude that the net positive suction head margin for the decay heat removal pumps as shown in the Final Safety Analysis Report will assure reliable operation in the recirculation mode.

7.0 INSTRUMENTATION AND CONTROL

7.1 General

In the Safety Evaluation Report, we noted that we would make a site visit to view the installation of equipment and verify the implementation of the design.

We visited the site in August 1977 and found the design implementation generally satisfactory. Several specific concerns were identified which were subsequently discussed with the applicant and satisfactorily resolved. We consider this aspect of the review completed.

7.3 Safety Features Actuation and Engineered Safety Features System

7.3.3 Feedwater and Main Steam Isolation

In the Safety Evaluation Report we noted that we would require additional information from the applicant to assure that, in the event of a main steam line break, the instrumentation and controls for the feedwater system isolation, and for main steam isolation if it is required, satisfy single failure and physical separation criteria.

The applicant has informed us that although the existing feedwater isolation valves are not safety grade, the present control systems conform essentially to our single failure and separation criteria. As noted in Section 15.2.2 herein, we consider the present system acceptable for operation during the first cycle. We will review the design of the isolation control system for operation after the first cycle when such information is submitted, as required in Section 15.2.2.

The applicant has shown that main steam isolation is not required during the first fuel cycle in the event of a steam line break accident. (See Section 15.2.2). If it is required after modifications are fully defined for operation after the first cycle, we will assure conformance with single failure and physical separation criteria.

7.4 Systems Required for Safe Shutdown

7.4.1 Emergency Feedwater Systems

In the Safety Evaluation Report, we found the emergency feedwater instrumentation, electrical, and control system design acceptable subject to final review of the steam line break analysis.

As stated in Section 15.2.2 herein, acceptable analyses of the steam line break have been submitted. Subject to the requirements imposed by that section, we conclude that the emergency feedwater instrumentation, electrical, and controls systems are acceptable.

7.5 Safety-Related Display Instrumention

In Supplement No. 1 to the Safety Evaluation Report, we noted that Babcock & Wilcox planned to test components identical to those used in this plant which are not similar to seismically qualified components.

The applicant informs us that performance of these tests is now uncertain. To supplement previous information, the applicant states that these components have been analyzed to verify their structural integrity during a seismic event, and that he judges that the floor response to which these instruments would be exposed in such an event would not cause failure. However, if this equipment should fail, portable meter readings could be obtained from the redundant and independent sensor outputs in the control room and at the shutdown panel outside the control room. Portable digital multimeters are available in seismically qualified housing in the control building cable room to monitor these functions. Based on the above, and on the information that all other instrumentation is similar to seismically qualified components, we conclude that for this plant the safety related display instrumentation is acceptable.

7.8 Qualification of Safety-Related Electrical Equipment

7.8.2 Environmental Qualification

In the Safety Evaluation Report, we noted that the applicant had stated that an analysis had been performed which showed that the vapor temperature in containment does not result in any safety-related equipment temperature in excess of the temperature for which that equipment was qualified.

As noted in Section 6.2.1 herein, revised analyses were performed to reevaluate the containment temperature profile and the effect of that temperature profile on safety-related equipment in the containment. We have not yet completed our review of these analyses. As further stated in Section 6.2.1, we will condition the operating license to require acceptable analyses prior to initial criticality.

In the Safety Evaluation Report, we further indicated that we would evaluate additional information requested of the applicant with respect to environmental qualification of balance-of-plant Class IE electrical equipment.

The applicant has provided acceptable documentation stating that all equipment involved has been certified to satisfy applicable industry standards.

Prior to startup following the first regularly scheduled refueling outage, we will require installation of an acceptable environmental temperature monitoring system to assure that the location of the Class IE equipment in buildings outside containment is maintained within the temperature range for which the equipment is designed to operate. For this time period, we conclude that the probability of an event requiring safety action coincident with loss of ventilation adversely affecting necessary safety equipment is sufficiently small that the incremental risk to the health and safety of the public is acceptable. We will condition the operating license appropriately in this matter, and consider this item resolved.

8.0 ELECTRIC POWER

8.1 General

In the Safety Evaluation Report, we noted that we would make a site visit to view the installation of equipment and verify the implementation of the design.

We visited the site in August 1977 and found the design implementation generally satisfactory. Specific concerns were identified which were subsequently discussed with the applicant and satisfactorily resolved. We consider this aspect of the review completed.

8.2 Offsite Power System

In the Safety Evaluation Report, we described the offsite power system, and noted that it satisfied General Design Criterion 17 and was therefore acceptable.

In the light of the events related to degraded grid conditions experienced at Millstone Unit 2, we had subsequently requested the applicant to provide additional information on the offsite power system for this facility. This information was submitted by the applicant and evaluated by us. This response increased our confidence in the ability of the system at Three Mile Island Unit 2 to withstand degraded offsite voltage conditions and avoid damaging interactions between the offsite and onsite emergency power systems. Shortly thereafter, we developed positions regarding these matters for all plants under review and all operating plants.

For this plant, we require that, prior to initial entry into operational Mode 1 (power operation), transformer tap settings be optimized for the full and minimum load conditions expected throughout the anticipated range of voltage variations of the offsite power source. We further require that prior to startup following the first regularly scheduled refueling outage, the following be acceptably implemented:

(1) Second level of under- or over-voltage protection with time delay,

(2) Appropriate interaction of onsite power sources with load shedding features, and

(3) Appropriate Technical Specifications and procedures covering onsite power source testing.

The applicant subsequently submitted adequate information to demonstrate that he will comply with these positions.

Given the ability of the present system to tolerate degraded grid or bus voltage conditions and the low probability of occurrence of such conditions, we conclude that these positions provide reasonable assurance that the health and safety of the public are adequately protected. We will appropriately condition the operating license to assure implementation of these positions.

8.3 Onsite Power Systems

8.3.1 Alternating Current Power System

We noted in the Safety Evaluation Report that for the system level safety features actuation signal there exists an indirect connection between redundant electrical trains by way of the output relays in the safety features actuation system cabinets. To allay this concern, the applicant had committed to provide additional documentation to demonstrate that these output relays connected in this manner would not degrade the safety features actuation system, and we stated we would report the results of our review of this documentation.

The applicant has provided this additional supporting information, and based on our review we conclude that the physical dimensions of these relays and the insulation resistance between contacts and between contacts and coil provide adequate isolation between redundant safety features actuation signal trains, and that this aspect of the existing design is acceptable. We consider this item resolved.

9.0 AUXILIARY SYSTEMS

9.2 Water Systems

9.2.1 Nuclear Services River Water Systems

Among the non-essential functions served by the nuclear services river water system is the cooling of the intermediate closed cooling water system through the intermediate closed cooling water heat exchanger. This heat exchanger, which forms part of the pressure boundary of the nuclear services river water system, is not identified in the Final Safety Analysis Report as being designed to seismic Category I requirements. However, failure of this heat exchanger in a seismic event could flood equipment important to safety, and could reduce flow available for essential equipment in the nuclear services river water system. Therefore, the intermediate closed cooling reactor heat exchanger should be designed to seismic Category I requirements.

The applicant has committed to completing the redesign of the intermediate closed cooling water heat exchanger and implementing the qualifications necessary to conform with seismic Category I requirements as soon as possible, but not later than entry into Mode 1 (full power operation). We find this commitment acceptable and will condition the operating license to require implementation of these modifications prior to that time.

Based on the low probability of a damaging seismic event during the short time interval between initial criticality and entry into Mode 1 and the applicant's commitment to design and modify the intermediate closed cooling water heat exchanger appropriately, we conclude that operation with this heat exchanger is acceptable.

9.5 Fire Protection Systems

In the Safety Evaluation Report, we indicated that additional requirements may be imposed on Three Mile Island Unit 2 to improve the capability of the fire protection system.

Subsequent to publication of the Safety Evaluation Report, we issued Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1976," which documents our position on fire protection for such plants as Three Mile Island Unit 2. We transmitted Appendix A to the applicant and requested performance of a fire hazard analysis and a reevalua tion of the fire protection program for this plant, including a comparison with Appendix A.

The applicant has submitted the required information, and our review, including an inspection of the plant, has been completed. Required improvements have been identified and the applicant has committed to implement these changes on an acceptable schedule, as listed below.

Hose Stations

Manual hose stations will be modified as necessary to conform to all the requirements of the National Fire Protection Association NFPA 14. Hose stretch tests will be performed and additional hose stations provided as necessary to reach any plant location in safety-related areas, and other areas containing major fire hazards, with at least one effective hose stream. Implementation is required prior to startup following the first refueling outage.

Fire Detectors in Control Room Cabinets and Consoles

The existing fire detectors in the control room will be relocated to provide quick detection of fire in each safety-related cabinet and console. Implementation is required prior to initial criticality.

Battery Room Hydrogen Monitor

Hydrogen monitors in the station battery rooms will be repositioned above the batteries and close to the ceiling for more effective and prompt detection of the hydrogen build-up. Implementation is required prior to initial fuel loading.

Automatic Water Suppression in Diesel Room Basement

An automatic water suppression system will be provided in each diesel generator room basement to suppress a possible fire in the basement and prevent damage to redundant safety-related cabling. Design features are required to be submitted for our review by March 1, 1978.

Openings Between Redundant Diesel Rooms

The opening in the common wall at the air intake area will be sealed with a three-hour concrete masonry structure. Ramped curbs will be added at the door openings in the common wall separating redundant diesel rooms to protect against a possible leakage of combustible oil. Implementation is required prior to startup following the first refueling outage.

Fire Doors, Frames and Hardware

Fire doors, frames, and hardware will be verified by the Underwriter's Laboratory certificate or an equal, for a fire rating commensurate with the barrier. Implementation is required by March 1, 1978.

Fixed Emergency Lights

Additional fixed, sealed beam emergency lights will be provided as necessary to facilitate emergency operations at remote shutdown panels and facilities. A minimum five-hour battery power supply will be provided for each of the emergency lights. Implementation is required by July 31, 1978.

Control Room Kitchen

All cooking and coffee brewing will be limited to the kitchen area. Deep frying will be prohibited. A lard and grease collecting filter over the range and an additional class B portable extinguisher will be provided, and the wooden cabinet removed from the area. Implementation is required prior to initial criticality.

Smoke Ejectors

Three fire fighter type smoke ejectors, with a combined capacity of at least 15,000 cubic feet per minute, will be provided to facilitate evacuation of toxic smoke from a fire area. Implementation is scheduled by July 31, 1978.

Fire Water Valve Supervision

All fire water valves will be either electrically supervised, locked open (or closed if appropriate), or provided with a tamper-proof seal and will receive periodic visual inspections. Implementation is required prior to initial fuel load.

Inspection of Outdoor Fire Hydrants

Procedures will be developed to provide for semi-annual inspection of all outdoor fire hydrants. Each hydrant will be inspected to ensure drainage of the dry barrel immediately prior to freezing weather, and reinspected to ensure proper operation immediately after the winter season. Implementation is required by May 1, 1978

Fire Fighters' Protective Clothing

Fire fighters' protective clothing, including helmet, coats, pants, gloves, and boots, will be provided for not less than eight men. Implementation is required by May 1, 1978.

Fire Fighting Plans

Fire fighting plans will be fully developed and documented to identify strategies for fighting fires in all safety-related areas, including the containment. Implementation is required by January 15, 1979.

Water Spray Protection

Water spray shields will be provided where both divisions of redundant safe shutdown equipment could be incapacitated by a rupture of fire water piping (as shown by analysis) or the application of fixed fire water. Adequate protection will be provided where both divisions of redundant safe shutdown equipment could be incapacitated by the application of manual fire water. Design features are required to be submitted for our review by March 1, 1978.

Diesel Oil Storage Area Overflow Protection

Modification will be made to prevent overflow of combustible liquid into the diesel rooms via ventilation openings in the event of a rupture of the diesel oil storage tanks. Implementation is required for May 1, 1978.

In the following areas, the applicant had not agreed with our positions or committed to implement the improvements required by these positions. We have considered the applicant's positions, and will require implementation of the following changes on the schedule noted.

Smoking in Safety-Related Areas

Smoking shall be prohibited in areas where the applicant's fire hazard analyses identify that a major fire could incapacitate safe shutdown of the plant. These areas shall include, but not be limited to the following eight areas: Fuel Handling Building (FA-007), Auxiliary Building (FA-009), River Water Pump House (FA-020), Control Building (FA-033), Cable Room (FA-045), Service Building and Control Building Area (FA-047), Reactor Building (FA-049), and the H&V Duct and Cable Tray Area (FA-041).

Smoking shall also be prohibited in the diesel oil storage area (FA-032) and the fuel handling building oil drum storage room (FA-008).

Implementation of these prohibitions is required prior to initial criticality.

Cable Spreading Room

An analysis shall be performed to demonstrate the capability of safely shutting down the plant independent of cabling and equipment in the cable spreading room. Completion of the analysis is required by May 1, 1978. Modifications necessary to achieve this capability shall be installed prior to startup following the first regularly scheduled refueling outage. In addition, the applicant shall provide either of the following prior to startup following the first regularly scheduled refueling outage:

(1) A manually operated fixed water system as a backup to the automatic Halon suppression system, or

(2) The automatic Halon suppression system, and acceptable fire retardant insulation around each cable tray not readily accessible by a manual fire hose stream, so that no fire may be expected to affect redundant safety trains. Conduit containing cables shall be similarly protected. Assurance shall be provided that cabling will not overheat.

Fire Door Supervision

All fire doors shall be electrically supervised with approximately two minute time-delayed alarms in a constantly manned area, or locked closed. The requirement for alarms may be waived for particular doors fitted with acceptable hold-open features designed to close due to a fire. A description of these features and a tabulation of such doors shall be provided by May 1, 1978. Implementation is required prior to startup following the first regularly scheduled refueling outage.

In addition to the above modifications, an incomplete item remains as discussed below. The applicant will complete the evaluations necessary to resolve these items in accordance with the schedule indicated. This schedule has been established such that should these evaluations require additional modifications, they will be implemented in a schedule consistent with completion of the above modifications.

Revised Fire Hazard Analysis

We require that the applicant's fire hazard analysis be revised to exclude the assumption that cables inside metal conduits will not be damaged by a design basis fire. Additional protection will be provided, as necessary, to preserve safe shutdown capability of the plant. This analysis is required to be completed by May 1, 1978.

Summary and Conclusions

We will appropriately condition the operating license to assure timely implementation of improvements and performance of analyses and evaluations.

The Technical Specifications have been modified to incorporate interim Technical Specifications which include limiting conditions for operation and surveillance requirements for existing fire protection systems and administrative controls. Following the implementation of the modifications of fire protection systems and administrative controls resulting from this review, the Technical Specifications will be similarly modified to incorporate the limiting conditions for operation and surveillance requirements for these modifications.

We plan to publish a separate fire protection Safety Evaluation Report describing our review more completely by the summer of 1978.

We conclude that significant steps are being taken to provide additional assurance that safe shutdown can be accomplished and the plant maintained in a safe condition during and after fire situations.

Based on our review, on the protection offered by the provisions of the interim Technical Specifications, and on the scheduled implementation of the above requirements, we further conclude that the fire protection program satisfies the objectives of the guidelines of Appendix A to Branch Technical Position 9.5-1 and meets General Design Criterion 3 and is therefore acceptable.

9.6 Diesel Generator Auxiliary Systems

9.6.2 Other Diesel Generator Auxiliary Systems

In the Safety Evaluation Report, we stated that each diesel generator is provided with two separate and independent compressed air starting trains, each including one air storage tank and each capable of providing five starts without recharging.

The applicant recently informed us that the design capability for each diesel generator, rather than for each train, is five starts. Actual tests indicate a minimum of seven starts are available. It is our position that for a two-train system, a design tank capacity of 10 starts per diesel generator is required. We will appropriately condition the license to assure implementation of this position prior to initial entry into Mode 1 (power operation). Based on the existing system capability, we conclude that fuel loading and low power operation prior to such implementation represents no undue risk to the health and safety of the public, and is acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.3 Main Steam Supply System

In the Safety Evaluation Report, we stated that the main steam isolation valves do not receive an accident signal. We also stated that we cannot verify the acceptability of the main steam supply system until the applicant submits appropriate steam line failure analysis.

At our request, the applicant has since modified the main steam isolation valves so that they are actuated by an accident signal.

As reported in Section 15.2.2 herein, acceptable analyses of the steam line break have been submitted. We therefore conclude that the main steam supply system is acceptable, subject to the requirements of that section.

10.5 Emergency Feedwater System

In the Safety Evaluation Report, we stated that the design of the emergency feedwater system was acceptable subject to final review of the steam line break analysis.

As stated in Section 15.2.2 herein, acceptable analyses of the steam line break have been submitted. Subject to the requirements of that section, we conclude tht the emergency feedwater system is acceptable.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.3 Gaseous Radioactive Waste Treatment System

The charcoal installed in the Engineered Safeguards System filter systems at Three Mile Island Unit 2 was reviewed and found acceptable under criteria which have since been upgraded by Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 1. These upgraded criteria are reflected in the plant Technical Specifications. A methyl iodide test of the installed charcoal for radioiodine removal efficiency showed a value of 96.79 percent compared to the Regulatory Guide requirement of 99 percent. A test for hardness showed a hardness number of 93 versus the Regulatory Guide value of 95.

Since the accident analysis doses reported in Section 15 of the Safety Evaluation Report, which are well within the guideline values of 10 CFR Part 100, are based on an assumption of 70 percent efficiency for organic iodide removal, and a test efficiency of 96.79 percent indicates an organic iodide efficiency of greater than 70 percent, the doses shown in the Safety Evaluation Report are conservative. The difference in hardness values is not considered significant and would have no discernable effect on the life or efficiency of the charcoal.

We therefore conclude that while the requirements of Regulatory Guide 1.52 with respect to charcoal characteristics represent an improvement in quality and performance, operation for the first fuel cycle with the presently installed charcoal is acceptable. We will requre installation of charcoal satisfying the requirements of the Technical Specifications prior to startup following the first regularly scheduled refueling outage in a condition of the operation license.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

Since issuance of the Safety Evaluation Report, revised information has been received from the applicant regarding the station staff organization. For completeness and accuracy, the second paragraph in this section in the Safety Evaluation Report should be replaced with the paragraph below. Our conclusions regarding the organizational structure and qualifications of personnel remain unchanged.

The station staff, under the direction of the Station Superintendent/Senior Unit Superintendent, is responsible for the operation of both units which are located at the site. The station staff consists of approximately 146 full-time employees functioning in four main groups: an operating and technical group for each unit (about 34 people in each group) responsible for plant operations and technical support in the areas of nuclear engineering, instrument and control engineering, mechanical engineering and electrical engineering; a maintenance group (about 60 persons) responsible for electrical, mechanical and instrument maintenance at the station; and a radiation protection group (about 15 persons) responsible for station radiation protection and chemistry programs.

13.6 Industrial Security

In the Safety Evaluation Report, we stated that the applicant had submitted a physical security plan applicable to Units 1 and 2, entitled, "Three Mile Island Industrial Security Program," as revised through Revision 9 dated July 15, 1975, and found it acceptable.

The applicant has since submitted a new physical security plan superseding the above and all subsequent revisions for Unit 2 only, entitled, "Three Mile Island Nuclear Station, Unit 2 (TMI-2) Interim Physical Security Plan," dated January 24, 1978, and Revision 1 to that plan dated January 27, 1978.

We conclude that this plan contains the necessary elements of security, is comparable to and consistent with Three Mile Island Unit 1, and includes the features and procedures currently required by 10 CFR Part 73.55, and is therefore acceptable. This plan is designated an interim plan since it will be further revised no later than August 24, 1978 to conform with all the requirements of 10 CFR Part 73.55.

14.0 INITIAL TESTS AND OPERATION

Our Safety Evaluation Report concluded that the initial test program proposed by the applicant was acceptable with the exception of the following two outstanding items:

(1) The proposed acceptance criteria for scram time testing of control rods.

(2) Test methods and techniques that would demonstrate the response times of protection system primary sensors and interfacing hardware between the measured variables and the input to the sensors for the reactor protection system and engineered safety features actuation system.

The applicant has since provided additional information on both these items which we have found acceptable. We therefore consider these matters resolved.

15.0 ACCIDENT ANALYSIS

15.2.2 Accidents

In the Safety Evaluation Report, we noted that various issues had arisen regarding steam line and feedwater line breaks, that analyses of these breaks were being prepared and reviewed, and that the adequacy of these analyses would be reported in a supplement to the Safety Evaluation Report.

During our subsequent review of these analyses, the staff position regarding equipment used to mitigate the consequences of a secondary system line break was defined as follows:

> For a spontaneous break anywhere in a main steam or main feedwater (secondary system) line, accident consequences must be mitigated only by safety grade equipment. However, for the assumed single failure in safety grade equipment, credit may be taken for non-safety grade equipment as a backup.
>
> For those portions of the secondary system where a break might be caused by a seismic event, accident consequences must be mitigated only by seismic Category I components, in accordance with General Design Criterion 2, after assuming a single failure in any seismic Category I component.
>
> If it can be shown that the unmitigated consequences of any secondary system line break accident on this plant are acceptable, considering the above positions and assuming a stuck rod and the availability or unavailability of offsite power, whichever is worse, then the present system design could be considered acceptable.
>
> Failing this, mitigating equipment must be upgraded, or new equipment added, to conform with the above position and assumptions, and appropriate analyses performed to show acceptable consequences.

Steam Line Break

The present system does not meet the above position in that equipment utilized to mitigate the consequences of a steam line break accident is not of appropriate safety grade and seismic classification. However, the applicant has committed to meet this position and to provide appropriate revised analyses. The applicant has further noted that it will not be possible to complete the installation of the required equipment by the expected time of the first fuel loading. Instead, he proposes to operate the plant with the presently installed equipment until the first refueling outage, at which time the appropriate modifications would be made. An action plan and schedule has been submitted covering design and analyses, review, equipment procurement, and installation of the modified system prior to startup following the first refueling outage.

The applicant has stated that interim operation prior to making the required modifications represents an insignificant risk to the health and safety of the public. To justify this interim operation, the applicant has submitted analyses of the steam line break accident during the first fuel cycle without credit for operation of any non-safety grade equipment to mitigate the consequences of that accident. He has also submitted discussions of the probability of the events in question and of the capability of the existing systems to function to mitigate the postulated accident.

In the applicants analysis covering the first fuel cycle, two periods during the transient were identified as critical relative to fuel damage. During the first period, from 0.8 seconds to five seconds after the break, a portion of the core departs from nucleate boiling until the control rods are inserted. Later in the transient, the continued cooldown of the moderator causes a return to subcritical power. Boron injection from the core flooding tanks terminates the loss of subcritical margin.

Conditions for departure from nucleate boiling early in the transient were maximized by assumptions regarding first fuel cycle reactivity feedback, initial steam generator inventory, break size and break location. Maximum runout feedwater flow and initiation of emergency feedwater at two seconds also act to maximize the rate of heat transfer from the primary system.

The reactor was assumed to be operating at 102 percent of full power at break initiation. Selection of the most adverse (beginning of first cycle) reactivity feed back parameters led to a slight power escalation. Reactor coolant pumps were assumed to start coasting down at break initiation. Analyses performed with the assumptions outlined above led to less than three percent of the fuel experiencing departure from nucleate boiling during the first five seconds.

The return-to-power period of the transient was analyzed both with reactor coolant pumps continuing to operate and with four pumps coasting down. End of first cycle and moderator temperature reactivity feedback parameters were assumed. Control rod worth was based on the minimum shutdown rod worth of two percent $\Delta k/k$ at hot shutdown conditions with the most reactive control rod stuck out. Conservative assumptions were made relative to boron injection from the core flooding tanks and high pressure injection pumps.

For the case of full reactor coolant flow, the minimum departure from nucleate boiling ratio was 2.09 occurring at 28 seconds. For the reactor coolant pump coast down case a minimum departure from nucleate boiling ratio of 1.93 occurred at 65 seconds. We conclude that no additional fuel would be expected to experience departure from nucleate boiling during the period of return to subcritical power.

An investigation was made to assure that fuel experiencing departure from nucleate boiling during the initial portion of the transient would not experience further damage during the return to power. It was determined that departure from nucleate boiling would occur in the upper portion of the rods during the early transient leading to peak clad temperatures of 1260 degrees Fahrenheit at eight seconds. The cladding temperature decreased thereafter due to convective cooling. Rewetting was not assumed. During the return to power, peaking occurred some six feet lower in the core with little or no effect on the previously heated region. No additional fuel damage would be expected during the return to power.

We agreed with the applicant that a coolable geometry would be retained for the fuel rods entering departure from nucleate boiling. The calculated peak cladding temperature of 1260 degrees Fahernheit coupled with the low differential pressure between the rod internal pressure and the primary coolant, precludes massive disruption or disintegration of the cladding due to oxidation embrittlement, bursting, or other potential failure mechanisms.

Three computer codes were used to accomplish the analyses. The PDQ07 code was used to determine reactivity feedback parameters. Systems analysis was performed with TRAP2. Core thermal and hydraulic calculations were performed with RADAR. Several iterative interactions were required to converge upon the results presented. The RADAR and PDQ07 codes have been reviewed and approved by the staff. The TRAP 2 code is currently under review by the staff. It is not expected that any method revisions which we may require will discredit the overall conclusions derived from these analyses.

The staff concludes that conservative analyses have demonstrated that less than three percent of the fuel will experience damage. The damage would not be expected to result in loss of coolable geometry. An independent evaluation by the staff concluded that the calculated radiological dose resulting from this postulated accident are within the guidelines of 10 CFR Part 100. (See Section 15.3.1) This determination applies to operation only during the first fuel cycle.

Long Term Cooling

The ability to achieve a long term cooling condition following a steamline break was investigated during the review. Detailed analysis of events through 75 seconds have indicated that conditions critical to fuel failure occur prior to that time. As noted above, the period from 0.8 to five seconds was the only time period for which fuel failure was predicted.

The applicant discussed conditions after 75 seconds, noting the following:

(1) High pressure injection would take the system water solid 10 to 15 minutes after the break in the absence of operator action.

(2) There is the possibility that natural circulation flow would be lost due to voiding in the reactor coolant piping.

It would be expected that for the period from 75 seconds to 125 seconds, the secondary system will continue to blow down and the primary system will continue to depressurize. With main steam isolation valve closure at 125 seconds, the unaffected steam generator is isolated and will begin to repressurize due to auxiliary feedwater and heat from the primary system. The repressurization will continue slowly until the secondary system safety valves begin to open. After 125 seconds, the primary system slowly reheats from around 400 degrees Fahrenheit at the rate of two to three degrees per minute. When the system becomes water solid at 10 to 15 minutes in the absence of operator action, the pressure will immediately rise to 2500 pounds per square inch gauge (safety valve set point) and continue to heat up. Both the primary and secondary systems will continue to release mass and energy from their respective safety valves until the operator terminates high pressure injection flow and the heat release from the secondary system exceeds the decay heat.

We have considered the possible loss of forced circulation and determined that no additional fuel failures are likely as the core will remain covered. Reactor coolant pump coastdown and circulation resulting from high pressure injection would also contribute to maintaining adequate cooling.

Maintaining primary system pressure at the relief valve set point value poses two concerns which are not of an immediate nature but will require additional study.

Since the vessel would be at a relatively low temperature when the system is repressurized, its toughness may be a concern. The applicant has confirmed to our satisfaction that the fracture toughness requirements of 10 CFR Part 50 Appendix G would not be exceeded in any steam line break transient during the first five years of operation.

The staff is proceeding with a generic review of reactor vessel pressure transient protection and pressurized water reactor core and primary coolant boundary response to a main steam line break. It is planned to complete these generic reviews by August 1979. On this basis, any additional requirements for plant modifications or changes in operating procedures would be identified at least three years before the applicant's conclusions regarding reactor vessel fracture toughness analyses would have to be reconsidered.

In addition, the steam generator tubes in the affected steam generator may be exposed to a differential pressure of 2500 pounds per square inch. The staff has considered the effect of this pressure on tube leakage and integrity, and through analysis and comparison with test results, has determined that increased leakage would not significantly affect our conclusions in Section 15.3.1 herein, and that a single through wall crack supplying the assumed one gallon per minute leak would not propagate to tube failure.

We conclude that long term cooling can be achieved following a steamline break. We expect to complete our generic review of vessel fracture toughness well in advance of any significant change in fracture toughness of the reactor vessel in this plant. If any changes are required as a result of this generic review, they will be required for this plant as appropriate.

Feed Line Break

The applicant has analysed events assuming loss of all feedwater and the case of a feedwater line break inside containment. A feedwater line break outside containment was considered the equivalent of the loss of feedwater event as check valves provided in each line would limit outflow.

The results of the analyses indicate adequate margin relative to potential fuel damage or overpressurization of the primary system.

A series of design changes have been made in the secondary system directed toward mitigating the consequences of a steamline break. Although the feedwater line breaks have not been reanalyzed reflecting these changes, the staff and applicant have reviewed the impact of such changes on the original analyses and have determined that the changes would not make the consequences more severe. Potential modifications to the secondary system to be implemented before the start of the second fuel cycle could affect the plant response to feedwater line breaks, especially if check valves are removed. The timing of isolation valves could also affect the consequences.

We have concluded that the feedwater line break analysis is acceptable and that system operation for the first fuel cycle is acceptable. Reanalyses of the modified system must be provided prior to the second fuel cycle.

In summary, we have considered the following relative to postulated secondary system line breaks:

- The design and quality of the presently installed system,
- The low probability of a damaging seismic event prior to the first refueling outage,
- The low probability of a random or seismically - induced secondary system line break,

- The probability of appropriate mitigative action of presently installed equipment in the event of a random or seismically - induced secondary line break,

- The applicant's analyses showing acceptable consequences of feed line breaks and an unmitigated steam line break during the first cycle, and

- The commitment of the applicant to make appropriate system modifications, and the action plan to implement this commitment.

Based on all the above, we conclude that the commitment by the applicant to make appropriate system modifications to meet our position regarding equipment to mitigate the consequences of postulated secondary system line break accidents is acceptable. We further conclude that short-term operation of the plant with the presently installed equipment represents no undue risk to the health and safety of the public, and therefore that operation of the facility need not be delayed pending implementation of the system modifications, and that interim operation of the plant prior to that implementation is acceptable. We will condition the operating license to require that system modifications and all required analyses be completed and acceptable prior to startup following the first regularly scheduled refueling outage.

15.3 Radiological Consequences of Accidents

15.3.1 General

Both we and the applicant have reevaluated the radiological consequences of a postulated steam line break accident during the first fuel cycle for Three Mile Island Unit 2. This accident is postulated to occur outside of containment upstream of the isolation valve. As a result of this event, the contents of the secondary side of the affected steam generator plus any primary-to-secondary system leakage is assumed to be vented and released to the atmosphere as an elevated release under fumigation conditions.

The applicant has indicated that, as a result of a stuck control rod, less than three percent of the fuel rods may be expected to fail if the accident were to occur during the first one-third of the fuel cycle. No fuel failures are predicted to occur if the accident were to occur after the first one-third of the cycle. Both we and the applicant conservatively assumed that this event resulted in the clad failure of three percent of the fuel rods and that a conservative estimate of the activity in the gap (as given by Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors") was assumed to be released and diluted within the primary system and transferred to the secondary side via the maximum leakage of one gallon per minute permitted by the technical specifications in the affected steam generator.

Since the secondary side of the steam generator remains dry for several hours prior to plant recovery, both we and the applicant assumed that all iodine transferred to the secondary side was released to the environment. Plant recovery is expected to occur within several hours after the event. Recovery would consist of a cooldown and depressurization of the primary system coupled with addition of feedwater to the secondary side of the steam generator. These procedures would eliminate the primary-to-secondary leakage and significantly attenuate the release of iodine transferred to the secondary side. The recovery procedure therefore terminates the accident. We have conservatively assumed that a total recovery is ineffective in preventing release of iodine until eight hours after the event.

The applicant concluded, as a result of his analysis, that the offsite thyroid dose was within the guideline values of 10 CFR Part 100. We performed an independent evaluation, using the acceptance criteria given in Standard Review Plan Section 15.1.5 (Appendix) that, for a steam line break coincident with a stuck rod, the doses should be within the values given in 10 CFR Part 100. Our assumptions for this evaluation were as discussed above and given in Section 15.3.2 herein. We calculated the doses to an individual located at the exclusion boundary for a two-hour period and an individual at the outer radius of the low population zone for the course of the accident (assumed to be eight hours). These doses are shown in Table 15.1 herein and are within the guideline values of 10 CFR Part 100.

Prior to startup following the first refueling outage, we require a revised analysis demonstrating that the radiological consequences for the revised system will be within the guideline values of 10 CFR Part 100.

15.3.2 Design Basis Accident Assumptions

Steam Line Break Accident During First Fuel Cycle

(1) Power = 2772 Megawatts thermal

(2) Three percent fuel with clad failure (with stuck control rod)

(3) 10 percent of iodine and noble gases fuel activity in gap

(4) Primary and secondary coolant activity concentrations as limited by Standard Technical Specifications (1.0 microcuries per gram I-131 equivalent and 100/E microcuries per gram noble gases for primary coolant and 0.1 microcuries per gram I-131 equivalent for secondary coolant)

(5) Primary-to-secondary leak rate as limited by the Standard Technical Specifications (1 gallon per minute)

TABLE 15.1

(PARTIAL)

POTENTIAL OFFSITE DOSES DUE TO STEAM LINE BREAK ACCIDENT DURING THE FIRST FUEL CYCLE

Accident	Two-Hour Exclusion Boundary (610 Meters)		Course of Accident Low Population Zone (3218 Meters)	
	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Steam Line Break	151	< 1	136	< 1

(6) 100 percent of iodine released to environment from secondary side.

(7) Release occurs as elevated release under fumigation conditions (according to Regulatory Guide 1.5)

0-2 hour χ/Q at exclusion boundary = 5.4×10^{-4} seconds per cubic meter
0-8 hour χ/Q at LPZ = 1.25×10^{-4} seconds per cubic meter

15.3.4 Fuel Handling Accident Inside Containment

The applicant has evaluated the consequences of a postulated fuel handling accident inside containment. The reactor building purge system was conservatively assumed to remain in operation exhausting to the atmosphere at 50,000 cubic feet per minute through charcoal and high efficiency particulate air filters, even though radiation monitors would be expected to isolate the containment after such an event. The plant systems intended to mitigate such an event consist of a seismic Category I exhaust system in the reactor building purge system, and seismic Category I filters. In his analysis, the applicant has taken credit for iodine removal by the filter systems. He concludes that the consequences of a fuel handling accident inside containment would be no greater than those for the same accident occurring in the spent fuel pool area.

We have independently evaluated the applicant's analysis and concur in this conclusion. We conclude that the consequences of a postulated fuel handling accident in containment would be well within the guideline values of 10 CFR Part 100, and that the existing plant systems provide effective mitigation, and are acceptable.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In Supplement No. 1 to the Safety Evaluation Report, we discussed the current status of each item on which the Advisory Committee on Reactor Safeguards had commented in their report of October 22, 1976.

Since Supplement No. 1, additional information on some of these items has become available, so that the status of these items is presently as follow:

(5) In Supplement No. 1 to the Safety Evaluation Report, we indicated that the applicant would submit and we would review prior to fuel loading the fire hazards analysis and reevaluation of the fire protection program for this plant relative to Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1976."

The applicant has submitted the required information, and our review, including an inspection of the plant, has been completed. Required improvements have been identified and the applicant has committed to implement these changes on an acceptable schedule. See Section 9.5 herein for additional information. We will appropriately condition the operating license to assure implementation of these improvements.

(6) With regard to hermetic seals on instrumentation and electrical equipment in containment during long-term post-accident operation, we had noted in Supplement No. 1 to this Safety Evaluation Report our understanding of the applicant's intentions in this matter.

The applicant has provided a list of instruments for which this concern is applicable and stated that all these devices are contained within gasketed enclosures. Maintenance and preventive maintenance procedures will be in accordance with the manufactuers recommendations, and will include replacement of the gasket whenever the enclosure is opened.

Since issuance of Supplement No. 1, the Committee has added this issue to their list of generic items as Item IID-2. In addition, the applicant has committed to resolve this issue for this plant in accordance with the generic resolution. As a generic issue, this matter will be dealt with on this plant and others when a final generic solution is developed.

(7) In Supplement No. 1 to this Safety Evaluation Report, we discussed some aspects of the Committee's concerns about non-essential loads interfering with the safety functions of the direct current power system.

At a subsequent meeting of the Committee, we noted that the direct current power system for this plant meets our present criteria and is similar to that of other plants recently licensed. As documented in a letter of April 26, 1977, R. F. Fraley to E. G. Case, subject: D.C. System Reliability, the Committee requested that the system be reevaluated as part of our generic evaluation of the reliability of such systems, as recommended in their letter of March 15, 1977, M. Bender to Lee V. Gossick, subject: Reliability of Power Supplies. If this evaluation results in changes applicable to Three Mile Island Unit 2, such changes will be considered at that time as appropriate.

(8) In Supplement No. 1, we had indicated that the hydrogen line passed only through a portion of the auxiliary building and that further evaluation of this situation would be made.

The original routing of the hydrogen line in the auxiliary building included a corridor which also contained cable trays involving both safety trains. This was deemed unacceptable, and the applicant has committed to a revised design. The 1/2 inch hydrogen line will be rerouted so that it is outside the auxiliary building except for an essentially vertical run down to the level of the makeup tank. In addition to the existing pressure controller and relief valve, the line will be protected by a rupture disc venting outside the building. Inside the building, the line will be continuously encased in a "guard pipe" which, in the event of a break in the inner line, will direct the hydrogen flow to atmosphere outside the building. Both the inner and outer pipes inside the building will be designed to seismic Category I requirements. These modifications will be implemented prior to fuel loading.

We find this solution acceptable and consider this item resolved.

(9) In Supplement No. 1 to this Safety Evaluation Report, we noted that the applicant would submit an analysis of the possible effects on plant controllability of instrument line failures.

In Amendment 53 to the Final Safety Analysis Report, the applicant presented a summary of his analysis of failure of various safety-related instrument lines. The tabular presentation included an indication of the effect of the line break on the affected instrumentation, on the plant, and on plant controllability, and an indication of how the failure would be diagnosed. None of the analyzed breaks appeared to present plant controllability problems of significance to public safety.

The question of instrument line failure has been identified by the Committee as one aspect of their general concern about system interactions. At a recent Committee meeting, we indicated that we consider the generic review of system interactions to be of high priority, and that the scope of our review is being defined. If any future evaluation of instrument line failure results in changes applicable to Three Mile Island Unit 2, such changes will be considered as appropriate.

(10) In Supplement No. 1 to this Safety Evaluation Report, we indicated we would review additional information to be submitted by the applicant regarding our concerns about the applicant's management organization as it concerns Three Mile Island Unit 2.

Additional information has since been submitted which satisfies our concerns regarding responsibility for and organization of station management. Section 13.1 of this Supplement provides additional information. We consider this item resolved.

20.0 FINANCIAL QUALIFICATIONS

In Supplement No. 1 to the Safety Evaluation Report, we noted that our review of the financial qualifications of the owners was based on the ownership arrangement in effect at the time. We further noted that changes in this arrangement were expected in the near future, and that we would review the effect of these changes on the financial qualifications of the owners in a later supplement.

We have since received the required information on the new ownership arrangement and have completed our review of this information. Metropolitan Edison will now receive a 25 percent share in the output from the Three Mile Island Unit 2 facility, Jersey Central 65 percent, and Pennsylvania Electric 10 percent.

Our conclusions as to the financial qualifications of the owners remains unchanged.

22.0 CONCLUSIONS

Based on our additional review as summarized herein, our conclusions as stated in the Safety Evaluation Report remain unchanged.

APPENDIX A

CHRONOLOGY

January 21, 1977	Letter from applicant on ECCS evaluation model
January 28, 1977	Letter from applicant re proposed draft tech specs
February 1, 1977	Letter to applicant requesting information by Systems Analysis Section
February 7, 1977	Letter to applicant acknowledging corrective and preventive actions
February 9, 1977	Letter from applicant transmitting Amendment 51
February 11, 1977	Letter to applicant on fire in motor control cooler
February 17, 1977	Letter from applicant concerning B&W ECCS reevaluation
February 17, 1977	Letter from applicant on schedule regarding steam line break accident analysis
February 25, 1977	Letter to applicant re guidance on implementing the new rule re physical security plan
February 28, 1977	Letter from applicant transmitting Amendment 52
March 15, 1977	Letter to applicant on secondary system line break
March 15, 1977	Letter to applicant requesting additional information to resolve certain open issues
March 18, 1977	Letter to applicant re fuel handling accident inside containment
March 24, 1977	Letter to applicant requesting additional information on proper selection of instrumentation trip setpoint values
March 25, 1977	Letter to applicant transmitting Supplement 1 to SER
March 25, 1977	Letter from applicant requesting Appeal Meeting

March 28, 1977	Letter from applicant requesting extension of construction permit
March 30, 1977	Letter from applicant transmitting Amendment 54
April 1, 1977	Letter from Shaw, Pittman, Potts and Trowbridge requesting amendment to construction permit
April 1, 1977	Memorandum and Order
April 11, 1977	Letter from applicant re vital power supply inverters
April 13, 1977	Letter to applicant re appeal meeting
April 13, 1977	Letter from applicant transmitting Amendment 55
April 22, 1977	Letter from applicant transmitting Amendments 55 & 56
April 26, 1977	Letter to applicant requesting additional financial information
April 27, 1977	Letter to applicant on reactor vessel overpressurization
May 2, 1977	Letter from applicant on fuel handling accident inside containment
May 5, 1977	Letter from applicant re steam line break accident
May 11, 1977	Letter from applicant re instrument trip setpoint values
May 25, 1977	Letter from applicant transmitting physical security plan
June 1, 1977	Letter from applicant re hermetic seals of instrument boxes
June 6, 1977	Letter to applicant re open issues
June 28, 1977	Letter from applicant re financial information
June 29, 1977	Letter from applicant re fire protection program
July 7, 1977	Letter from applicant re fire protection technical specifications
July 20, 1977	Letter from applicant transmitting Amendment 57
July 21, 1977	Meeting with applicant

August 1, 1977	Letter from applicant re fire protection technical specifications
August 1, 1977	Letter from applicant re irradiation of fuel rods
August 1-3, 1977	Electrical Site Visit
August 23, 1977	Letter from Shaw, Pittman, Potts and Trowbedge requesting amendment to construction permits to change ownership
August 26, 1977	Letter from applicant transmitting Amendment 58
August 29, 1977	Letter to applicant re fire protection
September 19, 1977	Letter to applicant re low grid voltage
September 19, 1977	Letter to applicant re physical searches of individuals
October 5, 1977	Meeting with applicant on steam generator instrumentation
October 6, 1977	Letter from applicant re reactor vessel supports adequacy
October 7, 1977	Letter from applicant transmitting Amendment 59
October 17, 1977	Letter from applicant submitting Fuel Densification Report
October 31, 1977	Letter from applicant transmitting Amendment 60
November 2, 1977	Meeting with applicant on Spray Pump NPSH
November 9, 1977	Meeting with applicant on open items
November 22, 1977	Meeting with applicant on steam generator sleeves
November 23, 1977	Letter from applicant re steamline break accidents
November 28, 1977	Letter to applicant re search requirements
December 8, 1977	Meeting with applicant on open items
December 9, 1977	Meeting with applicant on steam line break
December 12, 1977	Letter from applicant requesting extension of construction permit completion date

December 16, 1977	Letter from applicant transmitting Amendment 61
December 19, 1977	Initial Decision
December 19, 1977	Letter to applicant re fire protection review
December 22, 1977	Meeting with applicant on fire protection
December 28, 1977	Meeting with applicant on open items
January 3-6, 1978	Fire protection site visit
January 6, 1978	Letter to applicant on technical specifications
January 10, 1978	Meeting with applicant on steamline break
January 24, 1978	Letter from applicant transmitting Amendment 62

APPENDIX B

ERRATA

1.6 Modifications to Facility as a Result of Staff Review

In defining changes made in the plant design during our review in the Safety Evaluation Report, inadvertent ommission of the word "area" in two items may have been misleading. The correct wording follows.

17. Upgrading of control building area ventilation system to seismic Category I requirements (Section 9.4.7)

18. Restrictions on opening watertight doors between the turbine and control building area (Section 10.4)

7.0 Instrumentation and Control

8.0 Electric Power

As a result of a collating and publication error, Section 7.0, Instrumentation and Control, and Section 8.0, Electric Power, were inadvertently omitted from Supplement No. 1. These sections are reproduced on the following pages in their entirety.

7.0 INSTRUMENTATION AND CONTROL

7.2 Reactor Protection System

7.2.1 Description

In the fourth paragraph on page 7-2, the third sentence should read as follows, for clarity: The output signals of the averaging amplifiers are fed to an auctioneer device which selects the higest average power signal as an input signal to the integrated control system <u>and power range</u> recorder.

7.5 Safety Related Display Instrumention

In the Safety Evaluation Report, we noted that the need for readouts of certain display instruments which are not similar to seismically qualified components, and the seismic acceptability of the instrumentation, will be pursued further, and that physical separation in cable trays also remained a concern.

The display instruments not similar to previously qualified components are indicators and recorders used for reactor coolant and pressurizer temperature, and pressurizer and steam generator level. These components are mechanically idential to those which Babcock & Wilcox plans to test elsewhere by the summer of 1977. Should these tests indicate deficiencies in the components' seismic capabilities, we will require appropriate changes in Three Mile Island Unit 2.

With regard to cable trays, all those containing circuits associated with instrumentation for safe shutdown are designed to seismic Category I requirements. Most of these circuits are physically separated, and cable trays carrying these circuits contain only low voltage, low current instrumentation cables. Also, additional means such as diverse parameters are furnished to provide the necessary information.

We have reviewed the information provided by the applicant, including descriptions, cable tray layouts, and correlation data on instrumentation and associated cable trays. Based on this review, we conclude that the present design is adequate, subject to the testing discussed above. We consider this item resolved.

7.6 Other Systems Required for Safety

7.6.1 Changeover from Injection Mode to Recirculation Mode Following a Loss-of-Coolant Accident

We noted in the Safety Evaluation Report that we would review electrical schematics which the applicant agreed to supply showing the implementation of automatic changeover from injection to circulation mode.

The applicant has provided revised electrical schematics and diagrams. We have reviewed this information, and conclude that the electrical, instrumentation and control aspects of this modified design conform to appropriate requirements and are acceptable. We consider this item resolved.

7.8.1 Seismic Qualification

In the Safety Evaluation Report, we stated that, with the exception of the Report, the seismic qualification of seismic Category I electrical equipment is acceptable.

In Section 7.5 of this supplement, we conclude that the design of the safety related display instrumentation is seismically acceptable. Therefore, the exception in Section 7.8.1 is deleted, and we conclude that seismic quilification of the seismic Category I electrical equipment is acceptable.

7.9 Containment Electrical Penetrations

In the Safety Evaluation Report, we indicated that we had not yet reviewed the report covering prototype testing of the containment electrical penetrations.

We have since reviewed the information provided by the applicant concerning the electrical penetrations and conclude they conform to the applicable requirements and are acceptable. We consider this item resolved.

8.0 ELECTRIC POWER

8.3.1 Alternating Current Power System

We stated in the Safety Evaluation Report that we would report the results of our review of the electrical schematics to be provided by the applicant verifying implementation of certain requirements with regard to diesel generator trip devices.

The applicant has provided these electrical schematics and other information, which we conclude verifies that the design does conform with the staff position, and is acceptable. We consider this item resolved.

The Safety Evaluation Report further noted that the applicant had agreed to provide the capability to test periodically the undervoltage relays at the 4.16 kilovolt engineered safety feature buses, and to submit revised electrical schematics covering this feature.

These schematics and other information have been provided by the applicant and reviewed by us.

We conclude that the design is acceptable and consider this item resolved.