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Docket No. 50-320

Mr. John J. Barton
Acting Director of TMI-2
GPU Nuclear Corporation
P.O. Box 480
Middletown, PA 17057

Dear Mr. Barton:

The Nuclear Regulatory Commission has issued the enclosed Amendment of Order for the Three Mile Island Nuclear Station, Unit 2. This Amendment of Order changes the Recovery Mode Technical Specifications to reflect current conditions at the plant. The changed requirements reflected in the Proposed Technical Specification had been imposed by the Order of the Director of the Office of Nuclear Reactor Regulation on February 11, 1980. These changes are being made in response to your requests of February 23, March 18, and October 6, 1981. Additional changes to the containment building design pressure have also been made as discussed with GPU staff. This Amendment of Order is effective upon issuance. Changes in surveillance requirements in the Recovery Operations Plan as discussed with your staff are also approved.

Copies of the related Safety Evaluation and revised pages for the proposed Technical Specifications and for the Recovery Operations Plan are enclosed.

Sincerely,

Bernard J. Snyder, Program Director
Three Mile Island Program Office
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment of Order
2. Safety Evaluation
3. Proposed Technical Specifications Page Changes
4. Recovery Operations Plan Page Changes

cc w/encl:
See next page
L. King
J. Larson

OFFICE	TMIPO:NRR	TMIPO:NRR	TMIPO:NRR	ELD	NRR	NRR
SURNAME	TPoindexter;bg	RWeller	BSnyder	See previous concurrence	ECase	HDenton
DATE	3/15/82	3/17/82	3/ /82	3/ /82	3/ /82	4/ /82

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NUCLEAR REGULATORY COMMISSION

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of METROPOLITAN EDISON COMPANY, et. al. (Three Mile Island Nuclear Station, Unit 2)	))))))	Docket No. 50-320 OLA

AMENDMENT OF ORDER

1.

GPU Nuclear Corporation, Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (collectively, the licensee) are the holders of Facility Operating License No. DPR-73, which had authorized operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2) at power levels up to 2772 megawatts thermal. The facility, which is located in Londonderry Township, Dauphin County, Pennsylvania, is a pressurized water reactor previously used for the commercial generation of electricity.

By Order for Modification of License, dated July 20, 1979, the licensee's authority to operate the facility was suspended and the licensee's autnority was limited to maintenance of the facility in the shutdown cooling mode then in effect (44 F.R. 45271). By further Order of the Director, Office of Nuclear Reactor Regulation, dated February 11, 1980, a new set of formal license requirements was imposed to reflect the post-accident condition of the facility and to assure the continued maintenance of the current safe, stable, long-term cooling condition of the facility (45 F.R. 11282). Although these requirements were imposed on the licensee by the Director's Order of February 11, 1980, the TMI-2 license has not been formally amended. The requirements are reflected in the proposed Recovery Mode Technical Specifications presently pending before the Atomic Safety and Licensing Board. Hereafter in this Amendment of Order, the requirements in question are identified by the applicable proposed Technical Specification.

II.

By letters dated February 23, 1981, March 18, 1981, and October 6, 1981, the licensee proposed changes to the Recovery Mode Technical Specifications for Three Mile Island Unit 2 (TMI-2) to reflect current plant conditions.

Several categories of changes were proposed. The proposed changes are summarized as follows:

A) Deletion of Measurement Instrumentation and/or Surveillance Requirements for Equipment No Longer Required to be Operable.

1) The NRC staff finds that the present instrumentation used to monitor the shutdown condition of the reactor such as pressure and temperature transmitters, reactor coolant system boron sampling, and two operable source range neutron instruments is adequate; therefore, the deletion of the operability requirement for all differential pressure transmitters for the reactor coolant system as stated in Table 3.3-9 is acceptable as proposed.

2) The proposed Recovery Mode Technical Specifications eliminated requirements with respect to the Reactor Building Emergency Spray and Core Flooding Systems, but Table 3.3-10 of the Proposed Technical Specifications still contains operability requirements for the Core Flood Tank Level and Reactor Building Spray Pump Flow Instrumentation in addition

to associated surveillance requirements (Table 4.3.10). The staff does not find the subject instrumentation essential to maintaining the reactor coolant system in its present safe condition and therefore finds the proposed change acceptable.

3) Proposed Technical Specification 3.7.10.2 (d) and (e) requires that the Deluge System located at the Hydrogen Purge Exhaust Filter and the Reactor Building Purge Exhaust Filter be operable with automatic initiation of water spray to minimize the fire hazards associated with the charcoal filter train. The charcoal has since been removed, therefore the proposed deletion of the requirement to maintain the deluge systems in an automatic initiation status is approved. The deluge system would be isolated, thereby placing it in a manual mode and therefore preventing spurious activation signals. The fire detectors in the area are not affected and will continue to provide audible alarms.

4) The licensee has requested deletion of the requirements of proposed Technical Specification 3.3.3.8 which requires the fire detection instrumentation for the Balance of Plant Diesel to be operable as imposed in the February 11, 1980, Order. By letter dated April 28, 1980, the licensee proposed to delete these operability requirements for the BOP diesel generators and the 13.2Kv transmission line, replacing their capabilities with an existing 230Kv grid system. This modification was approved in the August 11, 1980, Modification of Order. Since the diesels are located outside of the restricted area fence, are not located near any vital equipment, and will no longer be required to perform a safety function, the NRC staff concurs with the licensee's proposal.

5) Proposed Technical Specification 3.6.1.3 (b) presently requires the containment air locks to be operable with an overall leakage rate of less than or equal to $0.05L_a$ at P_a, 56.2 psig. The licensee, however, has previously been granted an exemption from certain requirements of 10 CFR Part 50, Appendix J upon which Proposed Technical Specification 3.6.1.3 (b) is based. Therefore the licensee has requested and we have approved the deletion of the subject section. However, Section 3.6.1.3 (a) has not been deleted. The footnote indicating that surveillance requirement 4.6.1.3 (a) represents an exemption to Appendix J has also been deleted since Appendix J was recently modified (45 FR 62789) and now requires the same leak rate testing as is presently stated in Surveillance Requirement 4.6.1.3 (a).

B) Deletion of Systems or Structures No Longer Required to be Operable or Maintain Their Original Design Criteria.

1) The Proposed Recovery Mode Technical Specifications retained the Hydrogen Purge Cleanup System (proposed Technical Specification 3.6.4.3) to ensure the operability of the system in the event that purging of the containment building was approved by the NRC. On June 12, 1980, a Commission Memorandum and Order and a Temporary Modification of License were issued which authorized Metropolitan Edison Company to conduct a controlled purge of the containment building using that system with established off-site dose limits. The purge began on June 28, 1980 and was completed on July 11, 1980. Subsequent permission to perform periodic purges using another system, the Reactor Building Purge System, was granted by NRC letter NRC/TMI-80-119 from J.T. Collins to R.C. Arnold, dated July 31, 1980. The licensee has therefore requested that the

Hydrogen Purge Cleanup System, Section 3.6.4.3, be deleted from the proposed Technical Specifications and Surveillance Requirement 4.6.4.3, since its purpose was fulfilled upon the completion of the June 29, 1980, purge and the use of the system will not be required to maintain the containment building in a safe condition for the remainder of the Recovery Mode. Per the discussion in item (2) of the enclosed safety evaluation, the NRC staff finds that the deletion of Sections 3.6.4.3 and 4.6.4.3 acceptable.

2) Proposed Technical Specification 3.7.10.1 requires that at least 3 of 4 high pressure water pumps be operable to ensure that adequate fire suppression capability is available to confine and extinguish fires in any portion of the plant where safety related equipment is located. The licensee has proposed to modify this requirement so that only 2 of 4 high pressure water pumps would be required operable. The Fire Hazards Analysis for TMI determined that only 2 of 4 high pressure pumps were needed to provide a required combined capacity greater than 3575 gpm. The Bases for the Proposed Technical Specifications also reflect the same conclusion. Therefore, the staff has concluded that enough conservatism is present in the proposed Technical Specifications Section 3.7.10.1 to allow a change in the requirements to 2 of 4 high pressure water pumps being maintained in an operable condition without impairing the Fire Suppression System.

3) The licensee proposed to add occupational exposure considerations to the requirement stated in Section 5.2.2 of the Proposed Technical Specifications to maintain the 60 psig maximum internal pressure and 286 ^{o}F maximum internal temperature design. This would limit maintenance on the inside of the containment if radiological conditions posed a hazard to personnel. The staff approves this addition

but also sought to more correctly state a reasonable containment design pressure that should be maintained. The licensee performed an analysis evaluating the peak containment pressure under accident conditions. This analysis which was independently verified by the NRC staff, concludes that the maximum potential containment building pressure resulting from a LOCA is approximately 2 psig. Subsequent to the licensee's initial analysis, another evaluation was submitted by the licensee, (LL2-81-091, Barton to Snyder, 12/4/81), which concluded that it is conceivable that a fire inside of the containment building could increase the internal pressure to greater than 2 psig, causing a failure of the most limiting penetration (R-626). However, the fire analyzed by the licensee illustrated that even with the failure of penetration R-626, offsite releases would be less than 400 µCi with an average concentration at the station vent calculated to be 1.2×10^{-9} µCi/ml. The licensee's calculations and the staff's independent evaluation concludes that the releases would be well below the limits of 10 CFR 20 and within the scope of impacts assessed in the "Final Programmatic Environmental Impact Statement Related to the Decontamination and Disposal of Radioactive Wastes Resulting from the March 28, 1979, Accident at TMI-2", dated March 1981. Also to minimize the potential for a fire induced overpressure, the licensee has instituted fire hazards controls for the containment and is currently re-writing applicable procedures. In addition, piping and electrical penetrations other than R-626 are being redesigned to 5 psig. R-626 will remain as is until accessibility and personnel exposure permit its possible upgrading to the 5 psig value.

The staff's independent evaluation agrees with the licensee's conclusions and therefore Section 5.2.2 has been modified to state a containment design pressure of 2 psig. The design temperature will remain at 286 ^{O}F.

C) Addition or Modification of System or Structure Requirements.

1) The licensee had originally requested that the requirements of Section 3.6.1.4 of the Proposed Technical Specifications which limits maximum primary containment pressure to ≤ 0 psig be deleted. The basis for this request was that since the reactor building purge has removed most airborne contamination from inside containment, to the extent that normal leakage from the building will not affect the health and safety of the public, the requirement could be relaxed. However, based on discussions with the NRC staff on additional inside containment activities that could increase the airborne radioactivity concentration, the licensee withdrew the requested modification to increase the maximum operating containment pressure to + 1 psig per LL2-81-091, Barton to Snyder, dated December 4, 1981.

2) The proposed Technical Specifications do not require that the control room air inlet radiation monitor remain operable for all credible accident conditions. Presently, a failure of the control room air inlet radiation monitor could prevent the automatic shifting of the control room emergency air cleanup system to the recirculation mode in the event of an accident. The licensee has proposed to add the inlet monitor to proposed Technical Specification 3.7.7.1 and require it to be operable. The staff agrees that this change enhances the safety of control room personnel and concurs with the modification.

3) ANSI N18.1-1971, Section 4.4 is the currently referenced technical personnel qualification criteria in proposed Technical Specification 6.5.1.2 (c). Specified in Section 4.4 are reactor engineering and physics, instrumentation and control, radiochemistry, and the radiation

protection disciplines. The licensee has proposed to expand the academic requirements of the Plant Operations Review Committee (PORC) and the Generation Review Committee to a broader area as discussed in ANSI/ANS-3.1-1978, Section 4.7.2 The proposed change to the qualifications of the PORC and GRC members substitutes a basic requirement for a bachelor's degree plus three years of professional level experience in lieu of the existing requirement for only five years experience or, at most, a bachelor's degree plus one year of experience. The staff is in agreement that by using the proposed criteria the qualifications for PORC and GRC members will be higher than those presently required in the existing Proposed Technical Specifications and, accordingly we conclude that the proposed change is acceptable.

D) The Clarification of Ambiguity in Proposed Requirements.

1) The "action" statement of Proposed Technical Specification Section 3.9.12 (a) states that "with the fuel handling building/auxiliary building air cleanup system inoperable.....". However, it is not clear in this statement or statement 3.9.12 (b) that the fuel handling building and the auxiliary building air cleanup systems are independent of each other and that one of the two systems being inoperable should not affect any radioactive movements in the building solely associated with the other operating system. The staff concurs with a modification to 3.9.12 (a) and (b) as requested by the licensee to eliminate any ambiguity

2) Section 3.6.1.1 of the Proposed Technical Specifications presently contains an incorrectly used "and" which has the unintended effect of requiring that containment isolation be maintained when one containment isolation valve per penetration is open or inoperable by (a) at least one deactivated automatic valve secured in the isolation condition and

(b) at least one closed manual valve or blind flange. It was intended that method (a) or (b) would suffice when a penetration isolation valve was inoperable. Therefore, the licensee's request to so modify the specification is acceptable.

3) Paragraph 4.1.1.1.j. of the Surveillance Requirements presently indicates that water filled tanks, the surge tank and the degassed water supply tank shall be sampled to verify that they contain borated water. The proposed change by the licensee specifies what tanks are to be sampled by using more accurate system terminology and is, therefore, acceptable. Also, Section 4.1.1.1.j.2 specifies sampling to confirm a dissolved gas concentration of less than 15 scc/kg of water. An (*) and footnote has also been added as requested in order to more specifically state where the most representative sample should be taken for water being added to the reactor coolant system via the Standby Pressure Control System.

The staff's safety assessment of this matter is set forth in the concurrently issued Safety Evaluation. This evaluation concluded, in material part, that the amendment of order does not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the modified manner. Prior public notice of the amendment of order was therefore not required and the amendment of order is effective upon issuance.

It was further determined that the amendment of order does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. In light of this determination, it was concluded that the instant action is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5 (d) (4), that an environmental impact statement or environmental impact appraisal need not be prepared herewith.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, the Director's Order of February 11, 1980, is hereby revised to incorporate the deletions, additions, and modifications set forth in Attachment A hereto. For further details with respect to this action, see (1) Letter to B. Snyder, USNRC, from G. K. Hovey, Met-Ed/GPU, Technical Specification Change Request No. 26, dated February 23, 1981, (LL2-81-0049); (2) Letter to B. Snyder, USNRC, from G. K. Hovey, Met-Ed/GPU, Technical Specification Change Request No. 26, Addendum A, dated March 18, 1981, (LL2-81-0055); (3) Letter to B. Snyder, USNRC, from G. K. Hovey, Met-Ed/GPU, Technical Specification Change Request No. 26 Addendum B, dated October 6, 1981 (LL2-81-0229); (4) Modification of Order dated August 11, 1980; (5) Letter to R. C. Arnold, Met-Ed, from J. T. Collins, USNRC, granting permission to perform periodic purges using the Reactor Building Purge System, dated July 31, 1980; (6) Memorandum and Order dated June 12, 1980; (7) Order for Temporary Modification of License, dated June 12, 1980; and (8) the Director's Order of February 11, 1980.

All of the above documents are available for inspection at the Commission's

Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Commission's Local Public Document Room at the State Library of Pennsylvania, Government Publications Section, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: April 7, 1982
Dated at Bethesda, Maryland

References:

1. Letter to Bernard J. Snyder, NRC, from G. K. Hovey, Metropolitan Edison Company, "Technical Specification Change Report No. 26", LL2-81-0049, February 23, 1981.

2. Letter to Bernard J. Snyder, NRC, from G. K. Hovey, Metropolitan Edison Company, "Technical Specification Change Request No. 26, Addendum A", LL2-81-0055, March 18, 1979.

3. Letter to Lake Barrett, NRC, from G. K. Hovey, Metropolitan Edison Company, "Request for an Exemption from the Testing Requirements of 10 CFR 50, Appendix J", LL2-81-0094, May 11, 1981.

4. Letter to G. K. Hovey, Metropolitan Edison Company, from B. J. Snyder, NRC, "Appendix J Exemption Request", September 2, 1981.

5. Final Supplement to the Environmental Impact Statement for Unit 2, NUREG-0112, December 1976.

6. Fire Protection Program Evaluation, Three Mile Island, Unit 2, June 1977.

7. Final Programmatic Environmental Impact Statement, NUREG-0683, March 1981.

8. Letter to Bernard J. Snyder, NRC, from G. K. Hovey, Metropolitan Edison Company, "Technical Specification Change Request No. 26, Addendum B", LL2-81-0229, October 6, 1981.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GPU NUCLEAR CORPORATION

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2

Introduction

By letters dated February 23, 1981 (Reference 1), March 18, 1981 (Reference 2), and October 6, 1981 (Reference 8), the licensee proposed changes to the Recovery mode Technical Specifications, Appendix A/B for Three Mile Island Unit 2 (TMI-2). Because of the variety of types of changes requested, each has been separated and addressed individually. The requested changes to Appendix B were issued on May 6, 1981 in license Amendment No. 15; the requested changes to Appendix A are discussed herein.

Evaluation

(1) Instrumentation

A. The Reactor coolant flow indicators RC-14A-dPT1, RC-14A-dPT2, RC-14A-dPT3, RC-14A-dPT4, RC-14B-dPT1, RC-14B-dPT2, RC-14B-dPT3, and RC-14B-dPT4 are differential pressure transmitters located in hot legs A&B of the Reactor Coolant System. They are currently required to be operable in Table 3.3-9 of the proposed Technical Specifications. With the reactor coolant pumps operating, the pressure drop across a Gentile tube for each leg is measured and converted to a corresponding flowrate. The differential pressure/flowrate signal is then transmitted to the Reactor Protection System for reactor trip signal generation. This signal generation is presently not necessary since in the Order for Modification of License dated July 20, 1979, the licensee's authority was limited to maintenance of the facility in the present shutdown mode. The reactor coolant pumps which could produce the minimum required flow needed to utilize the indicators have not been used since the last pump was stopped on April 27, 1979 and natural circulation of the primary system was initiated. There is no present requirement or future need for using the reactor coolant pumps for core cooling during the remainder of the Recovery Mode. Therefore, the requirement to maintain these pumps in an operable status was deleted by an Amendment of Order dated November 14, 1980.

The licensee has proposed to modify Table 3.3-9 of the proposed Technical Specifications by deleting the operability requirement for all of the differential pressure transmitters used for reactor coolant system flow indication. Deletion of the associated Surveillance Requirement in Table 4.3-6 has also been requested. The staff finds that the present instrumentation used to monitor the shutdown condition of the reactor such as pressure and temperature transmitters, reactor coolant system boron sampling, and two operable source range neutron monitoring instruments adequately monitors the system and therefore finds the proposed changes acceptable.

B. The Reactor Building Emergency Spray and Core Flooding Systems were deleted from the proposed Technical Specifications in the February 11, 1980 Order. The licensee has therefore requested to delete from Table 3.3-10 of the proposed Technical Specifications the operability requirements of the Core Flood Tank Level and Reactor Building Spray Pump Flow Instrumentation and the associated Surveillance Requirements in Table 4.3.10. Based on the above discussion, we find that these instruments are not required to monitor the reactor coolant system in its present condition and therefore find the proposed changes acceptable.

(2) Hydrogen Purge Cleanup System

The February 11, 1980 Order retained the Hydrogen Purge Cleanup System (proposed Technical Specification 3.6.4.3) to ensure the operability of the system in the event that purging of the containment building was approved by the NRC. On June 12, 1980, a Commission Memorandum and Order and a Temporary Modification of License were issued which authorized Metropolitan Edison Company to conduct a controlled purge of the containment building with established off-site dose limits. The purge began on June 28, 1980 and was completed on July 11, 1980. Subsequent permission to perform periodic purges using another system, the Reactor Building Purge System, was granted by NRC letter NRC/TMI-80-119 from J. T. Collins to R. C. Arnold, dated July 31, 1980. The licensee therefore requested that the Hydrogen Purge Cleanup System, Section 3.6.4.3, be deleted from the proposed Technical Specifications and Surveillance Requirement 4.6.4.3. The staff has noted that the only credible event that could produce substantial amounts of hydrogen would be a recriticality accident concurrent with a temperature increase severe enough to cause the zircaloy cladding to begin decomposition. Recriticality was discussed in the Final Programmatic Environmental Impact Statement (PEIS) for TMI-2 issued in March 1981. Paragraph 4.1 of the PEIS states that "the most probable (although very unlikely) cause of recriticality was found to be boron dilution, which would be a slow enough process that any approach to criticality can be detected and remedied." This statement is still valid; therefore, the staff has concluded that this accident need not be designed against in reference to hydrogen production. Since the Hydrogen Purge System's purpose was fulfilled upon the completion of the June 28, 1980 purge and the use of the system will not be required to maintain the containment building in a safe condition for the remainder of the Recovery Mode, the Staff finds that the deletion of Sections 3.6.4.3 and 4.6.4.3 is acceptable.

(3) Fire Protection

A. Proposed Technical Specifications 3.7.10.2 d and e currently require that the Deluge System located at the Hydrogen Purge Exhaust Filter and the Reactor Building Purge Exhaust Filter be operable with automatic initiation of water spray. The reason for this requirement was to minimize the fire hazards associated with the charcoal in the filter train. The primary purpose for this charcoal was to remove gaseous iodine (I-131) from the effluent prior to release to the atmosphere. However, the iodine-131 levels in the containment building have been determined to be minimal. Therefore the charcoal has since been removed from both exhaust filters. With this removal, the fire hazard associated with these filters has also been eliminated. The licensee has therefore proposed that the requirement to maintain the associated deluge systems in an automatic initiation status be deleted. The deluge system would be isolated, thereby placing it in a manual mode rather than the previously required automatic mode. The staff agrees with Met-Ed that the manual mode would prevent spurious signals from needlessly activating the deluge system and since the fire detectors associated with this system are not affected, an alarm would still be provided in the event of fire in the filter area. Therefore, the staff approves the requested modification to proposed Technical Specifications 3.7.10.2 d and e.

B. The licensee has requested the deletion of the requirements reflected in proposed Technical Specification 3.3.3.8 which requires the fire detection instrumentation for the Balance of Plant Diesel to be operable as imposed in the February 11, 1980 Order. By letter dated April 28, 1980, the licensee proposed to delete the operability requirements for the BOP diesel generators and the 13.2Kv transmission line, replacing their capabilities with an existing 230 Kv grid system. This modification was approved in the August 11, 1980 Modification of Order. In that approval, the Safety Evaluation determined that the 230 Kv grid system adequately provided access to back up power capability for all of the plant equipment essential to the preferred reactor cooling modes. Since the diesels are located outside of the restricted area fence, are not located near any vital equipment, and will no longer be required to perform a safety function, the staff concurs with the licensee's proposal.

C. Proposed Technical Specification 3.7.10.1 requires that at least 3 of 4 high pressure water pumps be operable to ensure that adequate fire suppression capability is available to confine and extinguish fires in any portion of the plant where safety related equipment is located. The licensee has proposed to modify this requirement so that only 2 of 4 high pressure water pumps would be required operable. The Fire Hazards Analysis for TMI determined that only 2 of 4 high pressure pumps were needed to provide a required combined capacity greater than 3575 gpm (Reference 6). The Bases for the Proposed Technical Specifications also reflects the same conclusion. Therefore, the staff has concluded that enough conservatism is present in the proposed Technical Specifications Section 3.7.10.1 to allow a change in the requirements to 2 of 4 high pressure water pumps being maintained in an operable condition without impairing the Fire Suppression System.

(4) Containment Systems

A. Section 3.6.1.1 of the Proposed Technical Specifications presently contains a grammatical contradiction by requiring that with one containment isolation valve per containment penetration open or inoperable, maintain the affected penetration(s) closed, with either action (a) and action (b) taking place with no alternative designated. The staff agrees with the licensee that action (a) should be followed by "or" which will indicate two action alternatives as intended in this requirement.

B. Proposed Technical Specification 3.6.1.3 (b) presently requires the containment air locks to be operable with an overall leakage rate of less than or equal to $0.05L_a$ at P_a, 56.2 psig where L_a is the maximum allowable leakage rate (%/24 hrs) and P_a is the calculated peak containment internal pressure. The leakage requirements are set forth pursuant to the requirements of Title 10, Part 50, of the Code of Federal Regulations, Appendix J. These airlocks are installed as an integral part of the containment structure providing access to the reactor building while maintaining a barrier against the possible release of airborne contamination of the environment.

The licensee has requested a modification of the leak rate criteria (Proposed Technical Specification 3.6.1.3.b) because verification of compliance cannot be made without unacceptably high radiation exposures to personnel. In addition, the June 28, 1980 purge significantly decreased the airborne contamination levels, greatly reducing the concern over containment atmosphere leakage. Based in part on the above philosophy, the licensee has previously been granted an exemption from certain requirements of Appendix J upon which requirement 3.6.1.3 (b) is based. Therefore the proposal by the licensee to delete Section 3.6.1.3 (b) from the proposed Technical Specification and delete Section 4.6.1.3 (b) from the surveillance requirements is approved. However, Section 3.6.1.3 (a) has not been deleted. The footnote to surveillance requirement 4.6.1.3 (a) indicating that item (a) is an exemption to Appendix J has been removed. Since Appendix J was recently modified (45FR62789) and now requires the same leak rate testing on airlock doors as is stated the (*) and reference statement is not longer required.

C. Proposed Technical Specification 5.2.2 presently requires that the containment building be designed and maintained for a maximum internal pressure of 60 psig and a temperature of 286 ^{o}F. The licensee performed an analysis evaluating the peak containment pressure under accident conditions in support of a request for exemption from Appendix J (Reference 3). This analysis, which was independently verified by the NRC staff, concluded that the maximum potential containment building pressure was approximately 2 psig. Subsequent to the licensee's initial analysis, another evaluation was submitted by the licensee, (LL2-81-091), Barton to Snyder, 12/4/81), which concluded that it is conceivable that a fire inside of the containment building could increase the internal pressure to greater than 2 psig, causing a failure of the most limiting penetration (R-626). However, the fire analyzed by the licensee illustrated that even with the failure of penetration R-626, the offsite release would be less than 400 μCi with an average concentration at the station vent calculated to be 1.2×10^{-9} μCi/ml. The licensee's calculations and the staff's independent evaluation concludes that the releases would be well below the limits of 10 CFR 20 and within the scope of impacts assessed in the "Final Programmatic Environmental Impact Statement Related to the Decontamination and Disposal of Radioactive Wastes Resulting from the March 28, 1979, Accident at TMI-2", dated March 1981. Also to minimize the potential for a fire induced overpressure, the licensee has instituted fire hazards controls for the containment and is currently rewriting applicable procedures. In addition, piping and electrical penetrations other than R-626 are being re-designed to 5 psig. R-626 will remain as is until accessbility and personnel exposure permit its possible upgrading.

The staff's independent evaluation agrees with the licensee's conclusions and, therefore, Section 5.2.2 has been modified to state a containment design pressure of 2 psig. The design temperature will remain at 286 ^{o}F. A statement has also been added to Section 5.2.2 at the licensee's request, which states that any maintenance to be performed on the containment building to maintain this design pressure shall be done per occupational exposure considerations.

D. The maximum containment pressure was limited to <0 psig in proposed

Technical Specification 3.6.1.4 by the issuance of the February 11, 1980 Order. At that time airborne contamination levels were high and the required negative pressure value insured that all leakage would be into the building and not out. Since the Order was issued, the Reactor Building has been successfully purged. This decreased the airborne contamination levels to a value that normal leakage will not affect the health and safety of the public. Based on this discussion, the licensee has requested a modification of proposed Technical Specification 3.6.1.4 by increasing the internal pressure limitation to + 1 psig. In evaluating this proposal the staff agrees that the present airborne contamination levels inside containment are presently low. However, future operations that are expected to take place, such as containment building decontamination and the eventual removal of the reactor vessel head, have the potential for increasing the airborne radioactivity levels. With this increase, an outleakage from the building resulting from an internal positive pressure would be undesirable. The staff is of the opinion that because of this potential, the request to increase the pressure to a maximum of + 1 psig in proposed Technical Specification 3.6.1.4 is unacceptable and furthermore, unnecessary since the Reactor Building purge system can readily maintain a negative internal pressure by exhausting through HEPA filters. Exhaust flow through the HEPA filters will minimize the release of radioactive particles to the environment. This opinion was verbally transmitted to the licensee and subsequently the licensee has withdrawn the originally requested change per LL2-81-091, Barton to Snyder, dated December 4, 1981.

(5) Control Room Emergency Air Cleanup System

The proposed Technical Specifications do not require that the control room air inlet radiation monitor remain operable for all accident conditions. Presently, a failure of the control room air inlet radiation monitor could prevent the automatic shifting of the control room emergency air cleanup system to the recirculation mode in the event of an accident. The licensee has proposed to add the inlet monitor to proposed Technical Specification 3.7.7.1 and require it to be operable. The staff agrees that this change enhances the safety of control room personnel and therefore concurs with the modification.

(6) Personnel Qualifications

In accordance with the Proposed Technical Specifications for TMI-2, the plant Operations Review Committee (PORC) is composed of a Chairman, one member who meets or exceeds the qualifications of Regulatory Guide 1.8, September 1975, and seven members who meet or exceed the qualification requirements of Section 4.4 of ANSI N18.1-1971. In this change request, the licensee proposes to change the qualification requirements of the seven members from those specified in Section 4.4 of ANSI N18.1-1971 to those specified in Section 4.7.2 of ANSI/ANS 3.1-1978, which is a later version of ANSI N18.1-1971.

Section 4.4 of ANSI N18.1-1971 describes the qualification requirements of "Professional-Technical" personnel. Of these personnel, the responsible person in reactor engineering or physics was to have a bachelor's degree in engineering or the physical sciences and at least two years of experience in such areas as reactor physics, core measurements, core heat transfer, and core physics testing program. Other Professional-Technical personnel (in the areas of instrumentation and control, radiochemistry, and radiation protection) were to have at least five years of experience, of which two years should be related technical training. In each case, for these other Professional-Technical personnel, up to four years of the five years of experience could be fulfilled by related technical or academic training.

Section 4.7.2 of ANSI/ANS 3.1-1978 describes the qualifications of Staff Specialists who perform independent reviews of operational phase activities at nuclear power plants. These individuals are to have a bachelor's degree in engineering or the physical sciences as appropriate and three years of professional level experience in their respective fields of specialty. In special cases, eight years of experience in the specialty field may be acceptable without a degree. Also, credit toward experience may be given for advanced degrees in any of the specialized fields on a one-for-one basis up to a maximum of two years. The specialty fields covered are administrative control, nuclear power plant operations, nuclear engineering, metallurgy, quality assurance, non-destructive testing, chemistry and radiochemistry, instrument and controls, radiological safety, and mechanical and electrical engineering.

The proposed change to the qualifications of the PORC members thus would substitute a basic requirement for a bachelor's degree plus three years of professional level experience in lieu of the existing requirement for only five years experience or, at most, a bachelor's degree plus one year of experience.

Thus, we find that the proposed qualifications for the PORC members are higher than the qualifications required by the existing Technical Specifications and, accordingly, we conclude that the proposed change to the Technical Specifications is acceptable.

(7) Fuel Handling Building/Auxiliary Building Air Cleanup Systems

The "action" statement of Proposed Technical Specification Section 3.9.12 (a) states that "with the fuel handling building/auxiliary building air cleanup system inoperable....." However, it is not clear in this statement or statement 3.9.12 (b) that the fuel handling building and the auxiliary building air cleanup system are independent of each other and that one of the two systems being inoperable should not affect any radioactive movements in the building solely associated with the other operating system. Therefore, the staff concurs with a modification to 3.9.12 (a) and (b) as requested by the licensee to eliminate ambiguity.

Environmental Considerations

Based on the above evaluations, the approved changes in the proposed Technical Specifications will not result in any environmental impact beyond those considered in the Final Programmatic Environmental Impact Statement, NUREG-0683 (Reference 7) and the Final Supplement to the Environmental Impact Statement for Unit 2, NUREG-0112 (Reference 5). The staff has determined that these changes to the proposed Technical Specifications do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, the staff has further concluded that this modification of the proposed Technical Specifications involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5 (d) (4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this proposed Technical Specification change.

(8) Standby Pressure Control System

Paragraph 4.1.1.1.j.of the Surveillance Requirements presently indicate that water filled tanks, the surge tank and the degassed water supply tank, shall be sampled to verify that they contain borated water. The proposed change by the licensee specifies by using more accurate system terminology what tanks are to be sampled. Also Section 4.1.1.1.j.2 specifies sampling to confirm a dissolved gas concentration of less than 15 scc/kg of water. An (*) and a footnote has also been added as requested in order to more specifically state where the most representative sample should be taken for water being added to the reactor coolant system via the Standby Pressure Control System.

Conclusion

Based upon the staff's review of the proposed modifications to the proposed Technical Specifications, the staff finds that the licensee's changes are acceptable with the exception of modifications that were withdrawn by the licensee as discussed in part 4(D) of this safety evaluation. Based on the review of the licensee's approved requests, the staff has concluded that (1) the modifications do not authorize a significant change in the plant's operation; (2) the modifications do not involve a significant increase in the probability or consequences of accidents previously considered or a significant reduction in a margin of safety and therefore, does not involve a significant hazards consideration, (3) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the modified manner, and (4) such activities will be conducted in compliance with the Commission's regulations and the issuance of this Amendment of Order will not be inimical to the common defense and security or to the health and safety of the public.

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

Replace the following pages of Appendix "A" Proposed Technical Specifications with the enclosed pages as indicated. The revised pages contain vertical lines indicating the area of change.

Pages

3.3-8
3.3-9
3.3-11
3.6-1
3.6-2
3.7-4
3.7-5
3.7-7
3.9-1
B 3/4 6-2
5-1
6-3
4.1-2
4.3-7
4.3-8
4.6-1
6-6
4.6-2
4.6-3

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

DISPLAY METERS INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM OPERABLE CHANNELS
1. RC Inlet Temp	Cab. 217	50-650°F	1/Loop
2. Makeup Tank Level	Cab. 217	0-100 inches	1
DISPLAY INDICATIONS			
3. Control Rod Drive Trip Breakers	Cable Room	open/closed	1/Breaker

DISPLAY PATCH POINTS INSTRUMENT	PATCH POINT LOCATION	MEASUREMENT RANGE	MINIMUM OPERABLE CHANNELS
4. NI Intermediate Range Level Log N	Cab. 217	10^{-11} to 10^{-3} Amps	1
5. NI Source Range Level	Cab. 217	10^{-1} to 10^{6} CPS	1
6. Deleted			
7. Decay Heat Removal Flow	Cab. 217	0-5000 GPM	1/Loop
8. Steam Generator Pressure A/B	Cab. 217	0-1200 psig	1/St. Gen

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		MINIMUM CHANNELS OPERABLE
1.	Reactor Building Pressure	2
2.	Deleted	
3.	Reactor Coolant Outlet Temperature	2
4.	Steam Generator Level	1/steam generator*
5.	Borated Water Storage Tank Level	1
6.	High Pressure Injection Flow	1/loop
7.	Low Pressure Injection Flow	1/loop
8.	Deleted	
9.	Steam Generator Pressure	1/steam generator
10.	Incore Thermocouples	All available*
11.	Reactor Coolant System Pressure	1
12.	Reactor Building Water Level	1

*Report all failures pursuant to Specification 6.9.1.8.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

Instrument Location	Required Instruments Operable			Alternate Instrument		
	Heat	Flame	Smoke	Heat	Flame	Smoke
1. Auxiliary Building	N/A	N/A	2	N/A	N/A	2
2. Control Building						
351' Elevation	N/A	N/A	1	N/A	N/A	3
331' Elevation (Control Room)	N/A	N/A	1	N/A	N/A	1
305', 293', 280' Elevations	N/A	N/A	1	N/A	N/A	7
3. Control Building Area	N/A	N/A	1	N/A	N/A	3
4. Diesel Generator Building						
A Diesel	N/A	1	1	N/A	N/A	1
B Diesel	N/A	1	1	N/A	N/A	1
5. Fuel Handling Building	N/A	N/A	3	N/A	N/A	1
6. Reactor Building	N/A	N/A	1	N/A	N/A	1
RC-P-1A/2A	N/A	N/A	1	N/A	N/A	2
RC-P-1B/2B	N/A	N/A	1	N/A	N/A	2
7. River Water Pump House	N/A	1	1	N/A	N/A	2
8. Service Building	N/A	N/A	3	N/A	N/A	3
9. Deleted						

3.6 CONTAINMENT SYSTEMS

3.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained and all containment penetrations, including at least two OPERABLE containment isolation valves or a double barrier in each penetration, shall be closed when not required open per procedures approved pursuant to Specification 6.8.2. Purging or other treatment of the containment atmosphere shall be prohibited until approved by the NRC.

APPLICAILITY: RECOVERY MODE.

ACTION:

With one containment isolation valve per containment penetration open or inoperable, maintain the affected penetration(s) closed with either:

a. At least one deactivated automatic valve secured in the isolation position, or

b. At least one closed manual valve, or a blind flange.

CONTAINMENT AIR LOCKS

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for transit entry and exit through the containment, then at least one air lock door shall be closed unless otherwise specified per procedures approved pursuant to Specification 6.8.2.

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 56.2 psig. (Per occupational exposure considerations)

APPLICABILITY: RECOVERY MODE.

ACTION:

With an air lock inoperable, maintain at least one door closed and restore the air lock to OPERABLE status within 24 hours.

INTERNAL PRESSURE

3.6.1.4 Primary containment pressure shall be maintained between 0 psig and the minimum allowable pressure as determined from Figure 3.6-1.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the containment internal pressure outside the above limits, restore the internal pressure to within the limits within 1 hour.

AIR TEMPERATURE

3.6.1.5 Primary containment average air temperature shall not exceed 130°F.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the containment average air temperature greater than 130°F, reduce the average air temperature to within the limit within 24 hours.

3.6.2 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

3.6.4.1 One gas partitioner shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the gas partitioner inoperable, restore the inoperable instrument to OPERABLE status within 7 days.

HYDROGEN PURGE CLEANUP SYSTEM

3.6.4.3 Deleted.

3.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

3.7.7.1 The control room ventilation and emergecy air cleanup system shall be OPERABLE with:

a. Two control room supply fans and associated cooling coils,

b. Two control room bypass fans,

c. One charcoal adsorber and HEPA filter train, and

d. Two isolation dampers in the outside air intake duct.

e. The control room air inlet radiation monitor OPERABLE.

APPLICABILITY: RECOVERY MODE

ACTION:

a. With one control room supply fan or its associated cooling coil inoperable, restore the inoperable fan and/or cooling coil to OPERABLE status within 7 days.

b. With one control room bypass fan inoperable restore the inoperable fan to OPERABLE status within 7 days.

c. With the filter train inoperable, restore the filter train to OPERABLE status within 24 hours.

d. With one isolation damper in the outside air intake duct inoperable, restore the inoperable damper to OPERABLE status ro close the duct within 4 hours by use of at least one isolation damper secured in the closed position.

e. With the control room air inlet radiation monitor inoperable, restore it to OPERABLE status or place the control room emergency air cleanup system in the recirculation mode of operation within 4 hours.

3.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

3.7.10.1 The FIRE SUPPRESSION WATER SYSTEM shall be OPERABLE with;

a. At least 2 of the following 4 high pressure pumps shall be OPERABLE with their discharge aligned to the fire suppression header:

   1. Unit 1 Circulating Water Flume Diesel Fire Pump

   2. Unit 1 River Water Intake Diesel Fire Pump

   3. Unit 2 River Water Intake Diesel Fire Pump

   4. Unit 1 River Water Intake Motor Fire Pump

b. Two (2) separate water supplies of the following four (4) shall be available with at least 90,000 gallons each:

   1. Altitude Tank

   2. Unit 1 Circulating Water Flume

   3. Unit 1 River Water Intake Structure

   4. Unit 2 River Water Intake Structure

c. An OPERABLE flow path capable of taking suction from a water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser required to be OPERABLE per Specification 3.7.10.2 and 3.7.10.4.

APPLICABILITY: At all times

ACTION:

a. With 3 pumps or 3 water supplies inoperable, restore the inoperable equipment to OPERABLE status within 7 days or in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

DULUGE/SPRINKLER SYSTEMS

3.7.10.2 The Deluge and/or Sprinkler Systems located in the following areas shall be OPERABLE:

a. Diesel Generator Fuel Oil Tanks

b. Diesel Generator Building Air Intake

c. Air Intake Tunnel (Deluge - 2 of the 3 zones)

d. Hydrogen Purge Exhaust Filter AH-F-34#

e. Reactor Building Purge Exhaust Filters AH-F-31A/B#

f. Control Room Bypass Filter AH-F-5

g. Diesel Generator Rooms

h. Fuel Handling Building Exhaust Filter AH-F-14A/B#

j. Waste gas disposal filter WDG-F-1

k. Auxiliary Building exhaust filters AH-F-10A/B#

l. Condenser exhaust filters##

m. Auxiliary Building backup exhaust filters**

APPLICABILITY: At all times.

ACTION:

With one or more of the above required deluge and/or sprinkler systems inoperable, establish a roving (at least once perhour) fire watch* with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

* Except in the areas inaccessible per occupational exposure considerations.

**Supply lines may be isolated by one of four manually operated valves and by a manually operated deluge valve at the deluge station mounted on outside of building.

Supply line may be isolated by a single manually operated valve.

##Supply line may be isolated near the standpipe by two manually operated valves.

3.9 RADIOACTIVE WASTE STORAGE

FUEL HANDLING BUILDING/AUXILIARY BUILDING AIR CLEANUP SYSTEMS

3.9.12 The fuel handling building/auxiliary building air cleanup systems shall be OPERABLE with exhaust ventilation flow through the HEPA filters during system operation.

APPLICABILITY: At all times.

ACTION:

a. With one fuel handling building or auxiliary building air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days.

b. With no air cleanup system in the fuel handling building or the auxiliary building OPERABLE, suspend all operations involving movement of liquid and gaseous radioactive wastes in the affected building until the air cleanup system in that building is restored to OPERABLE status.

EPICOR II PROCESSED WATER

3.9.13 Discharge of water processed by the EPICOR II system shall be prohibited until approved by the NRC. Water processed by the EPICOR II system shall be discharged in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: At all times.

ACTION:

None except as provided in Specification 3.0.3.

REACTOR BUILDING SUMP WATER

3.9.14 Processing and discharge of water in the Reactor Building sump and Reactor Coolant System shall be prohibited until approved by the NRC. Water in the Reactor Building sump and Reactor Coolant System shall be processed and discharged in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: At all times.

ACTION:

None except as provided in Specification 3.0.3.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1.5 AIR TEMPERATURE

The average air temperature of the containment atmosphere is currently being maintained well below the 130°F limit. The action will maximize the service life of the instrumentation and equipment installed in the containment. Continued OPERABILITY of these items is required to continue monitoring and mitigating the March 28, 1979 incident.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The gas partitioner is provided to analyze the hydrogen concentration in the containment atmosphere so that actions can be initiated to reduce the hydrogen concentration if it approaches its flammable limit. If excessive hydrogen concentrations are detected, appropriate actions will be initiated to reduce the hydrogen concentration to a safe level.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area is shown on Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone is shown on Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-3.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 130 feet.

b. Nominal inside height = 157 feet.

c. Minimum thickness of concrete walls = 4 feet.

d. Minimum thickness of concrete roof = 3.5 feet.

e. Minimum thickness of concrete floor pad = 13.5 feet.

f. Nominal thickness of steel liner = 1/2 inches.

g. Net free volume = 2.1×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 2 psig and a temperature of 286°F. (Per occupational exposure considerations)

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Manager Radiological Controls who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Supervisor-Station Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Supervisor-Station Training and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The Plant Operations Review Committee (PORC) shall function to advise the Manager Site Operations on all matters related to nuclear safety and radioactive waste safety.

COMPOSITION

6.5.1.2 The Plant Operations Review Committee shall be composed of the:

a. Chairman - who shall have an academic degree in engineering or physical science field and a minimum of five years of applicable experience.

b. 1 Member - who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

c. 7 Members - who shall meet or exceed the qualification requirements of Section 4.7.2 of ANSI/ANS-3.1-1978.

The Manager Site Operations shall designate the Chairman and the Vice Chairman from among the members of the Plant Operations Review Committee.

BORON INJECTION (Continued)

h. At least once per 24 hours by verifying the BWST temperature is at least 50°F when the outside air temperature is less than 50°F.

i. At least once per 12 hours (when system is in operation) by verifying that the standby reactor coolant system pressure control system:

   1. Surge tank water volume is filled to between 55% and 80% of tank capacity and the tank is pressurized to the operating RCS pressure ± 25 psig but not higher than 600 psig.

   2. Isolation valves on the discharge side of the water filled tank nearest the reactor coolant system are open.

   3. The in-service nitrogen supply bank is pressurized to between 225 and 400 psig.

j. At least once per 7 days by verifying that the standby reactor coolant system pressure control system surge tanks and the charging water storage tank contain borated water with:

   1. A boron concentration of between 3000 and 4500 ppm.

   2. A dissolved gas concentration of less than 15 scc/kg of water.*

k. At least once per 31 days by verifying that the standby reactor coolant system pressure control system isolation valve on the discharge side of the water filled tank nearest the reactor coolant system closes automatically on a tank low level test signal.

*Dissolved gas concentration for the SPC System is determined by taking a representative sample from the sampling point located downstream of SPC-T-1.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

DISPLAY METERS INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. RC Inlet Temp.	M	R*
2. Make Up Tank Level	M	R*
DISPLAY INDICATIONS		
3. Control Rod Drive Trip Breakers	M	N/A
DISPLAY PATCH POINTS INSTRUMENT		
4. NI Intermediate Range Level Log N	M	R*
5. NI Source Range Level	M	R*
6. Deleted		
7. Decay Heat Removal Flow	M	R*
8. Steam Generator Pressure A/B	M	R*

*Nuclear detectors and all channel components located inside containment and components inaccessible due to occupational exposure considerations may be excluded from CHANNEL CALIBRATION.

TABLE 4.3-10

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. RC Inlet Temp.	M	R*
2. Make Up Tank Level	M	R*
DISPLAY INDICATIONS		
3. Control Rod Drive Trip Breakers	M	N/A
DISPLAY PATCH POINTS		
INSTRUMENT		
1. Reactor Building Pressure	S	R*
2. Deleted		
3. Reactor Coolant Outlet Temperature	S	R*
4. Steam Generator Level	N/A	N/A
5. Borated Water Storage Tank Level	S	R*
6. High Pressure Injection Flow	M**	R*
7. Low Pressure Injection Flow	M**	R*
8. Deleted		
9. Steam Generator Pressure	S	R*
10. Incore Thermocouples	S	R*
11. Reactor Coolant System Pressure	S	R*
12. Reactor Building Water Level (Heise Gauge)	N/A	SA#

*Nuclear detectors and all channel components located inside containment and components inaccessible due to occupational exposure considerations may be excluded from CHANNEL CALIBRATION.

**When applicable systems are in operation.

#Unless an appropriate engineering evaluation demonstrates the instrument to be in calibration.

4.6 CONTAINMENT SYSTEMS

4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

1. All accessible (per occupational exposure considerations) penetrations not required to be open per approved procedures during RECOVERY MODE are closed by valves, blind flanges, or deactivated automatic valves secured in their positions.

2. The equipment hatch is closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

CONTAINMENT AIR LOCKS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying less than or equal to 0.01 L_a seal leakage when the volume between the door seals is stabilized to a pressure to 10 psig.

b. At least once per 6 months by conducting an overall air lock leakage test at P_a, 56.2 psig, and by verifying that the overall air lock leakage rate is within its limit. (Per occupational exposure considerations)

INTERNAL PRESSURE

4.6.1.4 The primary containment internal pressure shall be determined to within the limits at least once per 12 hours.

AIR TEMPERATURE

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. RB nominal Elev. 350' (1 temperature indication)
b. RB nominal Elev. 330' (1 temperature indication)
c. RB nominal Elev. 305' (1 temperature indication)

4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

4.6.4.1 The gas partitioner shall be demonstrated OPERABLE at least once per 31 days by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance air.

b. Four volume percent hydrogen, balance air.

HYDROGEN PURGE CLEANUP SYSTEM

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10%.

   2. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

b. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ± 10%.

c. Deleted

6.5.2 GENERATION REVIEW COMMITTEE (GRC)

FUNCTION

6.5.2.1 The Generation Review Committee (GRC) shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear unit operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

f. Radiological safety

g. Mechanical and electrical engineering

h. Quality assurance practices

i. Radioactive waste operations

COMPOSITION

6.5.2.2 The GRC shall be composed of at least five members including the Chairman, appointed in writing by the Vice President Technical Functions. GRC members and alternates may be appointed from within the corporate organization or from external sources.

QUALIFICATIONS

6.5.2.3 Each GRC member and alternate shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years of technical experience, of which a minimum of three years shall be in one or more of the areas specified in 6.5.2.1. Nine years of appropriate experience in a specialized field is acceptable as an alternative to the above requirements. Credit toward experience may be given for advance degree work in above fields on a one-for-one basis up to a maximum of two years.

ALTERNATES

6.5.2.4 All alternate members shall be appointed in writing by the GRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in GRC activities at any one time.