August 24, 1999

Mr. James W. Langenbach, Vice President
and Director, TMI
GPU Nuclear, Inc.
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: ENGINEERED SAFEGUARDS FEATURE (ESF) SYSTEM LEAKAGE LIMIT AND CONTROL ROOM HABITABILITY EVALUATION (TAC NOS. MA4665 AND MA0246)

Dear Mr. Langenbach:

The Commission has issued the enclosed Amendment No.215 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) in response to your application dated February 2, 1999, as supplemented by letter dated July 29, 1999.

The amendment increases the allowable leakage for applicable portions of the ESF systems located outside containment and clarifies the system design requirements for the Auxiliary and Fuel Handling Building Ventilation System. As discussed with your staff and documented in your July 29, 1999, letter, the amendment is limited to Cycle 13 operation only. At least 6 months prior to the end of Cycle 13, we request that you resubmit your license amendment request along with the supporting control room habitability dose evaluation based on the staff's generic resolution of control room habitability concerns that we are working closely with the Nuclear Energy Institute Control Room Habitability Task Force to resolve.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Timothy G. Colburn, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No215to DPR-50
2. Safety Evaluation

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Docket No. 50-289

Enclosures: 1. Amendment No.215 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

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DATED: August 24, 1999

AMENDMENT NO. 215 TO FACILITY OPERATING LICENSE NO. DPR-50 THREE MILE ISLAND

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JERSEY CENTRAL POWER & LIGHT COMPANY

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GPU NUCLEAR, INC.

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 215
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission or NRC) has found that:

   A. The application for amendment by GPU Nuclear, Inc., et al. (the licensee) dated, February 2, 1999, as supplemented July 29, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 215, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

SBajwa

S. Singh Bajwa, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 24, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 215

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
iii	iii
3-62d	3-62d
4-45	4-45

TABLE OF CONTENTS

Bases

The Auxiliary and Fuel Handling Building Air Treatment System **(part of the Auxiliary and Fuel Handling Building Ventilation System - References 1 and 2), consists of four** banks of exhaust filters (AH-F2A, B, C, and D) and two sets of fans (AH-E-14A and C, and AH-E14B and D) which take the exhaust air from both the Auxiliary Building and the Fuel Handling Building and discharge it to the Auxiliary and Fuel Handling Building exhaust stack. **The** air **normally** passes through all **four filter banks when** either set of fans is in operation.

This system is not nuclear safety related. When available, it can be used to reduce the off-site dose releases; however, no credit was taken for this system in the analyses of the Waste Gas Tank Rupture (WGTR) Accident (Reference 4) or Maximum Hypothetical Accident (Reference 3), or for any other events releasing radioactivity through the Auxiliary Building. The dose consequences resulting from any of these events will be less than the 10 CFR 100 limits with or without system operation.

The in-place testing criteria for the HEPA and carbon adsorber banks, and the laboratory testing for the carbon adsorbers shall be performed in accordance with the test methods of ANSI/ASME N510-1980.

Note: The Fuel Handling Building ESF Air Treatment system controls the release resulting from a postulated spent fuel accident in the Fuel Handling Building per Technical Specification 3.15.4.

References

(1) **UFSAR Section 9.8.2 - "Fuel Handling Building Ventilation System"**

(2) **UFSAR Section 9.8.3 - "Auxiliary Building Ventilation System"**

(3) UFSAR Section 14.2.2.5 - "Maximum Hypothetical Accident"

(4) UFSAR Section 14.2.2.6 - "Waste Gas Tank Rupture"

4.5.4 ACCIDENT RECIRCULATION SYSTEMS LEAKAGE

Applicability

Applies to those portions of the Decay Heat, Building Spray, and Make-Up Systems, which are required to contain post accident sump recirculation fluid.

Objective

To maintain a low leakage rate from the accident recirculation systems to prevent significant off-site exposures.

Specification

4.5.4.1 The total maximum allowable leakage from the applicable portions of the Decay Heat, Building Spray and Make-Up System components as measured during refueling tests in Specification 4.5.4.2 shall not exceed 15.0 gallons per hour.*

4.5.4.2 During each refueling interval the following tests of the applicable portions of the Decay Heat Removal, Building Spray and Make-Up Systems shall be conducted to determine leakage:

a. The applicable portion of the Decay Heat Removal System that is outside containment shall be leak tested with the Decay Heat pump operating or by hydrostatic testing at no less than 350 psig, except as specified in "b".

b. Piping from the Reactor Building Sump to the Building Spray pump and Decay Heat Removal System pump suction isolation valves shall be pressure tested at no less than 55 psig.

c. The applicable portion of the Building Spray system that is outside containment shall be leak tested with the Building Spray pumps operating and BS-V-1A/B closed or by hydrostatic testing at no less than 350 psig, except as specified in "b".

d. The applicable portion of the Make-Up system on the suction side of the Make-Up pumps shall be leak tested with a Decay Heat pump operating and DH-V-7A/B open or hydrostatic testing at no less than 200 psig.

e. The applicable portion of the Make-Up system from the Make-Up pumps to the containment boundary shall be leak tested with a Make-Up pump operating or by hydrostatic test at no less than 3050 psig.

f. Visual inspection shall be made for leakage from components of these systems. Leakage shall be measured by collection and weighing or by another equivalent method.

***NOTE: This leak rate limit is only applicable for the Cycle 13 operating cycle.**

Bases

The leakage rate limit of 15 gph (measured in standard cold gallons) for the accident recirculation portions of the Decay Heat Removal, Building Spray, and Make-Up Systems is based on ensuring that potential leakage after a loss-of-coolant accident will not result in off-site dose consequences in excess of those calculated to comply with the 10 CFR 100 limits (Reference 1 and 2).

References

(1) UFSAR, Section 6.4.4 - "Design Basis Leakage" and Table 6.4-3 - "Leakage Quantities to the Auxiliary Building"

(2) UFSAR, Section 14.2.2.5(d) - "Effects of Engineered Safeguards Leakage During Maximum Hypothetical Accident"

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.215 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR, INC.

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated February 2, 1999 (which superseded an earlier submittal dated July 30, 1997), and supplemented by letter dated July 29, 1999, GPU Nuclear, Inc. (the licensee), requested changes to the Three Mile Island Nuclear Station, Unit 1 (TMI-1) Technical Specifications. The requested changes would increase the maximum allowable leakage of certain applicable portions of the engineered safeguards feature (ESF) systems located outside of containment, and clarifies the system design requirements for the Auxiliary and Fuel Handling Building Ventilation System (AFHBVS). The application also provided additional information related to the licensee's March 24, 1998, control room habitability dose assessment and superseded that submittal as well.

Specifically, the licensee requested the following:

1. TS Section 4.5.4 and its Bases be amended to (a) increase the maximum allowable limit from 0.6 gallon per hour (gph) to 15 gph for leakage from post-accident containment sump water recirculation ESF systems located outside of containment and (b) expand the number of ESF systems, which are assumed to leak during their intended operation in the course of a postulated loss-of-coolant accident (LOCA). Currently, the decay heat removal (DHR) system is specified as the ESF system that is assumed to leak. The licensee requested to add the building spray (BS) system and the makeup (MU) system to expand the ESF systems which are assumed to leak during their intended operation in the course of a postulated LOCA. The licensee also requested that this TS section be amended to revise the testing criteria to account for the highest pressure within that system during the containment sump recirculating phase.

2. TS Section 3.15.3 Bases be amended to clarify that no credit is taken for iodine removal by charcoal adsorbers in the AFHBVS in the licensee's radiological consequence analyses resulting from the waste gas tank rupture or maximum hypothetical accident.

In addition, the licensee also proposed to revise the table of contents to reflect the new title of TS Section 4.5.4 as "Engineered Safeguards Feature (ESF) Systems Leakage."

The proposed revisions will close specific open items identified in NRC Inspection Report 96-201, "Three Mile Island, Unit No. 1 Design Inspection," dated April 15, 1997, and the TMI-1 Licensee Event Report (LER) 97-004, Revision 0, dated April 4, 1997. These reports identified the following specific discrepancies between the TMI-1 TS and the TMI-1 updated Final Safety Analysis Report (UFSAR):

> "The TMI-1 UFSAR Section 6.4.3.6.4.4 and Table 6.4-3 stated the design basis leakage in the Auxiliary Building from the DHR system and BS system as 0.6 gph while the TMI-1 TS Section 4.5.4, Decay Heat Removal System Leakage, allowed 6 gph leakage from the DHR system. There were no TS for BS system leakage."

To rectify the discrepancy, the licensee proposed an interim license amendment in a letter dated September 19, 1997, to reduce the maximum allowable leakage limit for the ESF systems outside containment in TS Section 4.5.1 from 6 gph to 0.6 gph to conform with the licensing-basis leakage stated in the UFSAR. The licensee also committed to limit the total leakage from post-accident containment sump water recirculating portions of the DHR system as well as the BS and MU systems. The staff accepted the proposed interim license amendment and issued License Amendment No. 205 on October 15, 1997. Therefore, the proposed TS revision supersedes the licensee's request granted by License Amendment No. 205 and closes specific open issues identified in the Nuclear Regulatory Commission (NRC) inspection report and the licensee's LER.

By letter dated July 29, 1999, the licensee per agreement during a July 27, 1999, telephone call with NRC staff reviewers and management, submitted a supplemental change to the proposed license amendment dated February 2, 1999, to limit the applicability of this amendment request to the next operating cycle (Cycle 13). This did not change the staff's initial no significant hazards consideration determination or the FEDERAL REGISTER notice.

2.0 EVALUATION

To demonstrate the adequacy of the TMI-1 ESF systems designed to mitigate the radiological consequences of the design-basis accidents with the increased ESF system leak rate of 15 gph and without relying upon the charcoal adsorbers in the AFHBVS for removal of iodine, the licensee reevaluated the offsite and control room radiological consequences resulting from the most limiting LOCA. The licensee included the results of these offsite and control room dose calculations in the submittals.

In the submittals, the licensee concluded that the existing ESF systems at TMI with the increased ESF leak rate and without relying upon the charcoal adsorbers in the AFHBVS will still provide assurance that the radiological consequences at the exclusion area boundary (EAB) and the low population zone (LPZ) resulting from the postulated LOCA will be within the dose reference values specified in 10 CFR Part 100.

To review the licensee's radiological consequence assessments, the staff performed a confirmatory dose calculation for the following three potential fission product release pathways following the postulated LOCA:

(1) containment leak
(2) post-LOCA leakage from engineered safety features systems outside containment
(3) post-LOCA leakage from engineered safety features boundary valves to the borated water storage tank (BWST) vented to the environment

The results of the staff's radiological consequence calculation are given in Table 1 (Attachment 1). The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Tables 2 through 4 (Attachments 2 through 4 respectively).

2.1 Containment Leak Pathway

The radiological consequences resulting from containment leakage following a postulated design basis LOCA were evaluated. The staff reviewed the licensee's analysis and performed a confirmatory dose calculation. The staff's calculation incorporates the appropriate assumptions of the regulatory positions in Regulatory Guide 1.4. The staff used a containment leak rate of 0.1 percent per day based on the TMI-1 TS limit for the first 24 hours and a 0.05 percent per day leak rate for the remaining 29 days. No iodine removal credit was taken for the AFHBVS filters.

In the event of a postulated LOCA, the containment spray system is activated to reduce the post-accident energy and to remove airborne iodine. The licensee assumed a mixing rate of 4.9 unsprayed volumes per hour (58,000 cfm) between the sprayed and unsprayed portions of the containment atmosphere based on the operating capacity of two out of three containment emergency cooling fans. To be conservative, the licensee claimed elemental iodine removal by the containment spray for the first 28.5 minutes during initial spraying of fresh chemical injection solution, and no iodine removal credit was taken during the recirculating phase of the containment sump solution. The staff concludes that these assumptions are acceptable.

2.2 Post-LOCA Leakage from Engineered Safety Features Systems

In the TMI-1 UFSAR, the licensee assumed 0.6 gph for the post-LOCA leakage from ESF systems in its radiological consequence calculations. The maximum allowable leakage limit from the ESF components is also currently specified in the TMI-1 TS as 0.6 gph. In this license amendment, the licensee requested to change the 0.6 gph limit currently specified in the TMI-1 TS and the 0.6 gph value assumed in the UFSAR to 15 gph.

The staff and the licensee assumed (1) 30 gph of ESF leakage (twice the amount of the TS limit) for the entire duration of the accident (30 days) and (2) a gross failure of a passive component to occur at a rate of 50 gpm starting 24 hours into the accident and lasting for 30 minutes in accordance with the Standard Review Plan (SRP). The staff assumed that 6.4 percent of the iodine contained in the ESF leakage water becomes airborne based on the containment sump water pH and the initial sump water temperature. The airborne iodine is assumed to be released immediately to the environment.

2.3 Post-LOCA Leakage Pathway from BWST Vent

The licensee identified another potential fission product release pathway from ESF system leakage following a postulated LOCA. This pathway is leakage through boundary valves to the BWST, which is vented to the environment. The licensee estimated the leakage rate to the BWST to be 180 gph for the first 5 hours, reduced to 102 gph until 24 hours, and reduced to 96 gph for the remaining 29 days. The licensee stated that the180 gph leakage value is based on the capability of leakage-detection tests, and the remaining leakage values are estimated. The staff accepted the proposed leakage values. All of the ESF system leakage reaching the BWST is assumed to be in the liquid form. The staff assumed an iodine partition factor of 100. No credit is taken for iodine plateout in the BWST. The staff allowed a volatile iodine mixing efficiency of 10 percent in the BWST air space for holdup and decay prior to release to the environment.

2.4 Control Room Habitability

By letter dated August 14, 1986, the staff issued a supplemental safety evaluation (SE) regarding NUREG-0737, Item III.D.3.4, "Control Room Habitability." In the SE, the staff required the licensee to address only the whole body and skin doses stating that the thyroid dose for the control room operator would be deferred until completion of the staff's accident source term reevaluation. In a letter dated September 24, 1997, the staff requested that the licensee submit analyses to show that the thyroid dose to the control room operator is in compliance with General Design Criterion (GDC) 19 acknowledging that the source term study and its implementation at operating plants had not been completed. The licensee submitted the control room operator thyroid dose calculation in a letter dated March 24, 1998, and supplemented it by this proposed TS revision.

The requirements for the protection of the control room operators under postulated accident conditions are specified in GDC 19. The licensee has proposed to meet these requirements by incorporating shielding and emergency ventilation systems in the control room and by having an adequate supply of self-contained breathing apparatus in the control room. Upon receipt of an engineered safeguards signal or a high radiation level in the control building ventilation system return air duct, the control room is isolated by automatic closing of the air inlet and exhaust dampers. This places the control building emergency ventilation system (CBEVS) in a recirculating mode with 39,000 cfm of control room air being circulated through redundant high-efficiency particulate air filters and charcoal adsorber units. Following the isolation of the control room, outside air can be brought into the CBEVS by modulating outside air intake and exhaust dampers to maintain a positive pressure in the control room.

To be conservative, the licensee assumed that both outside air intake and exhaust air dampers, have failed to close fully at their intermediate positions, resulting in leakage of outside air through both dampers into the control room. The licensee measured in-leakage through the inlet air damper as 3400 cfm and in-leakage through the exhaust air damper (backflow) as 600 cfm. The 600 cfm value was based on the lower limit of detection of the air flow measurement instrument. The licensee assumed that the failed dampers can be repaired within 24 hours. In addition, the licensee assumed 10 cfm of unfiltered in-leakage for ingress/egress for the entire duration of the postulated LOCA (30 days).

The results of the staff's control room radiological consequences are given in Table 1 (Attachment 1). The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Tables 2 and 3 (Attachments 2 and 3). Thyroid dose to the control room operator calculated by the licensee and the staff are slightly higher than the dose acceptance criteria specified in the SRP. The SRP interprets that the appropriate acceptance criterion for thyroid dose to comply with GDC 19 is 30 rem.

Currently, the staff is closely working with Nuclear Energy Institute (NEI) Control Room Habitability Task Force to resolve generically all control room habitability related issues. Therefore, because the thyroid dose is not significantly exceeded and we expect to resolve generic control room habitability issues in the near future, we conclude that the requested license amendment is acceptable for the Cycle 13 operating cycle. At the end of the cycle, the licensee must resubmit the requested license amendment along with the TMI-1 control room habitability evaluation based on our generic resolution with the industry.

2.5 Atmospheric Relative Concentrations at the Exclusion Area Boundary, Low Population Zone and for the Control Room

By letter dated October 15, 1998, supplemented by another letter on February 3, 1999, the licensee submitted a proposal to amend the atmospheric dispersion (X/Q) values for the EAB and the LPZ for the radiological consequence assessments resulting from the postulated design basis accidents. The staff reviewed the licensee's analysis and concluded that the proposed revision of the X/Q values was acceptable. These revised X/Q values, which are used in this evaluation, are listed in Table 3 (Attachment 3).

The licensee calculated the 95th percentile X/Q values for a postulated release from the containment to the control room through the yard intake using the diffuse source release option of the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). This option assumes that the release occurs over an area rather than from a single point. As the distance between the release location and receptor increases, the calculational results of these two methodologies converge. Following discussions with the staff, the licensee agreed to recalculate the initial diffusion coefficients by dividing the assumed release area width and height by 6. Initially, the licensee's values were estimated by dividing the area width and height by 4 and 2, respectively. Onsite meteorological data for 1992, 1993, 1995, and 1996 were used as input in these and all of the following calculations.

Since the exact location and dimensions of releases from the containment building are not known, the staff also made comparative calculations assuming a point release for multiple points around the circumference of the containment. The shortest distance between the containment building and yard intake is approximately 90 meters. The X/Q values calculated making this assumption were about 75 percent higher than those calculated assuming the diffuse source release option. The X/Q values calculated by the staff for a point at an average release distance were only about 20 percent higher than those calculated using the diffuse source release option. It would be very conservative to assume that the release will occur at the point on the containment closest to the yard intake. In addition, in the event of an accident, effluents could be released from more than one location (e.g., leaking from two penetrations) or over a small area.

The control room ventilation exhaust is not a point at which air would normally be drawn into the control room. However, the ventilation exhaust is very near the containment building and, in the event of damper failure in the air exhaust duct, air adjacent to the containment building could be drawn by backflow into the control room. The licensee estimated X/Q values for this pathway by assuming releases at locations scattered around the containment surface. The licensee examined test data of flow around other building configurations and applied the results to the TMI building complex. The tests did not model the TMI site itself. The licensee estimated surface concentration coefficients (K values) by applying the test data to the TMI site and calculated X/Q values by assuming fractional releases from the selected locations with mixing in the building wake cavity. This method was selected because of concerns in using the ARCON96 methodology, a Gaussian plume model, for a postulated release location very near the receptor location.

Since the exact location and dimensions of releases from the containment building are not known, the staff made comparative calculations assuming a point release for multiple points around the circumference of the containment. Atmospheric dispersion (X/Q) values calculated by the staff using the ARCON96 methodology for a point at an average release distance were about 2 to 5 times higher than the licensee's calculated X/Q values. In the worst case, if effluents were released directly across from the ventilation exhaust, with limited mixing, the resultant X/Q values could be much greater than the weighted X/Q values calculated by licensee. However, it is unlikely that the release would occur at a point on the containment closest to the ventilation exhaust. In addition, in the event of an accident, effluents could be released from more than one location (e.g., leaking from two penetrations) or over a small area. Further, the ventilation exhaust is not the design intake and, therefore, would not routinely draw in air in event of an accident. Backflow would only occur under conditions of damper failure.

The licensee has proposed use of X/Q values for a release from the borated water storage water tank to the control room via the control room ventilation exhaust instead of the yard intake. The borated water storage tank is closer to the ventilation exhaust than to the yard intake, although in a different wind direction. The staff confirmed that this assumption is more conservative than assuming intake at the yard intake. The licensee calculated X/Q values for postulated releases from the auxiliary building to the yard intake by assuming a ground level release from the auxiliary building at a point close to the yard intake. The staff performed confirmatory calculations.

The staff finds the X/Q values calculated by the licensee to be acceptable for use in the dose assessment for the Cycle 13 operating cycle. The NRC is attempting to resolve control room issues generically through an industry forum and the staff may reassess the control room X/Q values proposed by the licensee in this license amendment request, if needed, based on conclusions drawn from resolution of the control room generic issues.

The staff has reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequence resulting from the postulated LOCA. The doses calculated by the staff are listed in Table 1 (Attachment 1). The staff's analysis confirms the licensee's conclusion that the radiological consequences would not exceed the dose reference values specified in 10 CFR Part 100 for the EAB and LPZ. For the control room operator dose, the staff is closely working with NEI Control Room Habitability Task Force to resolve generically all control room habitability related issues. Therefore, we conclude that the requested technical specification changes are acceptable for the Cycle 13 operating cycle. At the end of the cycle, the licensee must resubmit the requested license amendment along with the TMI-1 control room habitability evaluation based on our generic resolution with the industry. Therefore, the staff has determined that the license amendment requested by the licensee to increase the maximum allowable ESF system leakage limit is acceptable for operating cycle 13.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 14283). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachments:
1. Table 1, Radiological Consequences
2. Table 2, Assumptions Used in Computing Radiological Consequence
3. Table 3, Atmospheric Dispersion Factors (X/Q) Values
4. Table 4, Control Room X/Q Values

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TABLE 1

Radiological Consequences
(rem)

Release Pathways	EAB		LPZ		Control Room	
	Thyroid	WB[1]	Thyroid	WB[1]	Thyroid	WB[1]
Containment Leak	145	2	91	<1	10	<1
ECCS Leak	127	<1	152	<1	18	<1
BWST Vent	<1	<1	11	<1	7	<1
TOTAL	272	2	254	<1	35	<1
Acceptable Dose Criteria	300	25	300	25	30	5

[1] Whole body

Table 2
Assumptions Used in Computing Radiological Consequences

Parameter	Value
Power level, MWt	2825
Fraction of core inventory released, fractions	
Noble gases	1.0
Iodine	0.25
Iodine chemical forms, fractions	
Organic	0.04
Elemental	0.91
Particulate	0.05
Primary containment leakage, %/day	
0 to 24 hours	0.1
1 to 30 days	0.05
Primary containment free volume, ft^3	2.16E+6
Sprayed volume	1.45F+6
Unsprayed volume	7.10E+5
Containment spray	
Flow rate, gpm	2,500
Average drop fall height, ft	96
Spray solution pH	>8.0
Iodine removal rate by spray, $hour^{-1}$	
Elemental	10
Particulate	6.06
Organic	0
Sump water recirculating startup time, minutes	28.5
ECCS leak rate, gph	
0 to 24 hours, gph	15
24 to 24.5 hours, gpm	50
24.5 to 720 hours, gph	15

Table 2
Assumptions Used in Computing Radiological Consequences
(Cont'd)

Sump water volume, ft^3	6.3E+4
Iodine partition factor	6.4
Sump water leakage to BWST, gpm	
0 to 5 hours	3
5 to 24 hours	1.7
24 to 720 hours	1.6
BWST volume, ft^3	4E+4
BWST air mixing efficiency, %	10

Table 3
Atmospheric Dispersion Factors (χ/Q Values)
(sec/m^3)

0-02 hour EAB	8.0E-4
0-02 hour LPZ	1.4E-4
2-08 hour LPZ	6.0E-5
8-24 hour LPZ	3.9E-5
1-04 day LPZ	1.6E-5
4-30 day LPZ	4.0E-6

Table 4 Control Room χ/Q Values (sec/m^3)

Yard Intake (Normal)

0-2 hour	3.40E-4
2-8 hour	2.25E-4
8-24 hour	1.02E-4
1-4 day	7.61E-5
4-30 day	4.99E-5

Ventilation Exhaust Intake (back flow)

0-8 hour	1.96E-3
8-24 hour	1.37E-3
1-4 day	9.14E-4
4-30 day	5.09E-4

Borated Water Storage Tank Vent

0-2 hour	8.45E-4
2-8 hour	5.23E-4
8-24 hour	2.49E-4
1-4 day	1.77E-4
4-30 day	1.19E-4

Auxiliary Building Release

0-2 hour	3.02E-3
2-8 hour	2.08E-3
8-24 hour	1.02E-3
1-4 day	6.63E-4
4-30 day	4.37E-4