October 2, 2001

Mr. Mark E. Warner
Vice President - TMI Unit 1
AmerGen Energy Company, LLC
P.O. Box 480
Middletown, PA 17057

SUBJECT: TMI-1 - ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE RE: CONTAINMENT INTEGRITY DURING REFUELING (TAC NO. MB1051)

Dear Mr. Warner:

The Commission has issued the enclosed Amendment No. 236 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated January 23, 2001, as supplemented August 22 and September 17, 2001.

The amendment revises the TMI-1 Technical Specification (TS) requirements for containment integrity associated with the personnel and emergency air locks during fuel movement and refueling operations. Partial implementation of an alternate source term (AST) in accordance with Regulatory Guide 1.183, "Alternate Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," and Title 10 of the *Code of Federal Regulations*, Section 50.67, which you had also requested in your application, was not necessary because the Commission approved full implementation of an AST for TMI-1 in Amendment No. 235, dated September 19, 2001.

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,

/RA/

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

Docket No. 50-289

Enclosures: 1. Amendment No. 236 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

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Accession No. ML012710214 *SE Received. No substantive changes made.

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DATE	10/1/01	10/01/01	4/5/01	9/26/01	10/2/01	10/2/01

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AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 236
License No. DPR-50

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

   A. The application for amendment by AmerGen Energy Company, LLC (the licensee), dated January 23, 2001, as supplemented August 22 and September 17, 2001, 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. 236, are hereby incorporated in the license. The AmerGen Energy Company, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, contingent upon the licensee's implementation of regulatory commitments contained in the licensee's letters dated August 22 and September 17, 2001, and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Lakshminaras Raghavan, Acting 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: October 2, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 236

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
3-44	3-44
3-45	3-45
--	3-45a
4-29	4-29

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 236 TO FACILITY OPERATING LICENSE NO. DPR-50

AMERGEN ENERGY COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated January 23, 2001, as supplemented August 22 and September 17, 2001, AmerGen Energy Company, LLC (the licensee), submitted a request for approval of changes to the Three Mile Island Nuclear Station, Unit 1 (TMI-1), Technical Specifications (TSs). The August 22 and September 17, 2001, letters provided additional clarifying information which did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice (66 FR 31702). The licensee also provided camera-ready TS pages by letter dated September 25, 2001.

The requested changes revise the TMI-1 TS requirements for containment integrity associated with the personnel and emergency air locks during fuel movement and refueling operations. Specifically, the licensee is proposing to revise TS Sections 3.8.6, 3.8.7, and 4.4.1.3, to allow personnel and emergency air locks to remain open during fuel movement and refueling operations and complete containment closure in 45 minutes following a fuel handling accident (FHA). To support this request, the licensee also requested approval to use selective implementation of an alternate source term (AST) in accordance with Regulatory Guide (RG) 1.183, "Alternate Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," dated July 2000, and Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident Source Term," for the TMI-1, Fuel Handling Accident Inside Containment design-basis accident analysis.

2.0 BACKGROUND

The regulations in 10 CFR Part 50, Appendix A, Criterion 61, "Fuel storage and handling and radioactivity control " require fuel storage and handling, radioactive waste, and other systems which may contain radioactivity to be designed to assure adequate safety under normal and postulated accident conditions. Additionally, 10 CFR 50.67 and RG 1.183, allow licensees to revise their design-basis accidents as identified in their safety analysis report, and provide guidance on acceptable full or selective implementations of alternative source terms. The licensee has proposed to selectively implement an alternative source term in accordance with 10 CFR 50.67 and RG 1.183 in re-analyzing TMI-1's limiting design-basis FHA inside containment. The proposed change would allow the licensee to revise its TMI-1 TS requirements for containment integrity associated with the personnel and emergency air locks and other penetrations during fuel movement and refueling operations. Approval of the licensee's request for selective implementation of an AST was not necessary because the Commission approved full implementation of an AST for TMI-1 in Amendment No. 235 dated September 19, 2001.

Specific TS changes reviewed by the Nuclear Regulatory Commission (NRC) staff include:

- A revision to TS Section 3.8.6 and 3.8.7, proposed to allow the personnel and emergency air lock doors and other penetrations to remain open during fuel movement and refueling operations. An additional footnote to TS Sections 3.8.6 and 3.8.7 proposed to identify the requirement for administrative controls associated with closure of the personnel and emergency air lock doors and other mechanical penetrations.

- A revision to TS Section 3.8.7 proposed to provide equivalent isolation methods for other penetrations consistent with NUREG-1430, "Babcock & Wilcox Owners Group (B&WOG) Standard Technical Specifications (STS)," Section 3.9.3.c.1, dated April 1995.

- The addition of TS Section 3.8.11 to specify the requirement to maintain at least 23 feet of water over the top of the reactor vessel flange and the actions required if this level is not maintained.

- A revision to TS Section 4.4.1 to annotate that the surveillance check to determine operability of the personnel and emergency door interlocks shall allow for the opening of both doors as provided for in TS 3.8.6.

The Bases of TS Section 3.8 is also revised to provide a description of the plant conditions under which the personnel and emergency air locks and other penetrations may be open during fuel movement and the administrative controls applicable to these conditions. This Bases is further revised to provide the description of equivalent isolation methods for other penetrations consistent with the B&WOG STS.

The NRC staff has previously approved similar license amendment requests for Vogtle Electric Generating Plant, Units 1 and 2, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, and Surry Power Station, Units 1 and 2, to permit containment penetrations to remain open during fuel movement and refueling operations.

3.0 EVALUATION

3.1 Revised TS Section 3.8.6

Current TS Section 3.8.6 requires that at least one door in each of the personnel and emergency air locks be closed during fuel loading and refueling operations. The licensee has proposed to revise TS Section 3.8.6 to require at least one door in each of the personnel and emergency air locks be capable of being closed during fuel loading and refueling operations.

The current TSs are consistent with the current analysis of the FHA inside containment. The allowance to have containment personnel and emergency air lock doors open with direct access from containment atmosphere to the outside atmosphere (unisolated) during fuel movement and CORE ALTERATIONS is based on: (1) confirmatory dose calculations of a revised FHA as approved by the NRC staff which indicate acceptable consequences, and (2) commitments from the licensee to implement acceptable administrative controls which would ensure that in the event of a refueling accident (even though containment closure is not required to meet acceptable dose consequences) that the open personnel and emergency air lock doors can and will be promptly closed following containment evacuation.

The licensee, in its January 23, 2001, letter, stated that administrative controls would be in place to ensure that personnel are aware that air lock doors are open and that specific individual(s) are designated and available to close the air lock doors as part of a required evacuation of containment. Additionally, if any obstructions are in the path of the air lock doors that could prevent closure of an air lock door they will be capable of being quickly removed. In an August 22, 2001, response to an NRC staff request for additional information (RAI), the licensee provided clarification of its administrative controls. Based on the clarification of these administrative controls, the licensee committed to (1) track the status of all penetrations through licensee procedure 1101-3, “Containment Integrity and Access Limits,” which ensures a means to isolate them if necessary; (2) revise Procedure 1505-1, Data Sheet 4, “Low Pressure Containment Boundary Closure Device Status,” to ensure continued awareness by shift operating/refueling crews of the open status of penetrations; and (3) revise Procedure 1101-3 to designate qualified personnel and provide and stage tools and equipment to ensure containment closure within 1 hour.

The licensee’s proposed containment closure time of 1 hour following an FHA did not comport with the recommended closure time of ½ hour in RG 1.183. In a letter dated September 17, 2001, in response to staff concerns of not providing sufficient bases for a 1-hour containment closure, the licensee committed to a containment closure time of 45 minutes. The 45-minute closure time will provide reasonable assurance of prompt containment closure following an FHA even though containment closure is not required to meet acceptable dose consequences. The NRC staff finds the administrative controls acceptable because they satisfy the intent of Technical Specification Task Force (TSTF) traveler 68, “Containment Personnel Airlock Doors Open During Fuel Movement,” and RG 1.183, since the personnel and emergency air lock doors will be capable of being closed.

General Design Criterion (GDC) 64, “Monitoring Radioactive Releases,” requires monitoring effluent discharge paths during normal operation, anticipated operational occurrences and accidents. In the August 22, 2001, RAI response, the licensee stated that the reactor building purge system will be in operation, such that any leakage through the personnel and emergency air lock doors will be from the environment into the reactor building because the purge system maintains the reactor building at a negative pressure. The licensee indicated that should the purge system become inoperable, samples will be collected via installed portable air samplers located in the reactor building. Results of samples collected from the portable air samplers in conjunction with conservative estimates of air flow out of the building, would be used to determine effluent releases. In addition, the licensee has committed to revising Procedure 1103-1, to provide for placement of continuously operated particulate and radioiodine air sampling equipment inboard open air locks during core alteration and fuel handling activities. The licensee complies with the requirements of GDC 64 and the proposed TS Section 3.8.6 change is, therefore, acceptable.

3.2 Revised TS Section 3.8.7

The current TS Section 3.8.7 states:

> During the handling of irradiated fuel in the Reactor Building, each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
>
> 1. Closed by an isolation valve, blind flange, or manual valve, or
>
> 2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

The licensee proposed to revise TS Section 3.8.7 as follows:

> During the handling of irradiated fuel in the Reactor Building, each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
>
> 1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or capable of being closed, or
>
> 2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

The current TSs require that penetrations be closed by an isolation valve, blind flange, manual valve, or be capable of being closed by an operable automatic containment purge and exhaust isolation valve. The licensee is proposing to revise this requirement to require that the penetrations be capable of being closed by the appropriate valves or other equivalent methods. Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. Administrative controls ensure that (1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and (2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

As discussed in Section 3.1, above, the licensee stated that appropriate administrative controls would be maintained for penetrations that provide direct access to the atmosphere. In addition, (1) Procedure 1101-3 will be revised to ensure a means to isolate penetrations; (2) Procedure 1505-1 will be revised to ensure continued awareness of the status of penetrations by shift operating/refueling crews; and (3) Procedure 1301-2, Data Sheet 3, “Checks Applicable at Cold Shutdown & Refueling Shutdown,” requires checks to confirm that containment closure is available for all penetrations. Further, the licensee stated that it uses gasketed swing-away blind flanges with internally fire-foamed [resistant] penetrations as an equivalent method to close a penetration. In a letter dated August 22, 2001, in response to an NRC staff RAI, the licensee stated that in the event that timely containment closure is required, the fire-foamed penetration alone would provide this function. The licensee also stated that an internal engineering change request is written to evaluate any equivalent method selected and would be the basis for any procedure change(s) addressing the equivalent method.

The staff finds the administrative controls acceptable because they are in general agreement with TSTF 312, “Administrative Control Containment Penetrations,” and RG 1.183 because the penetrations will be capable of being closed within 45 minutes.

GDC 64 requires monitoring effluent discharge paths during normal operation, anticipated operational occurrences, and accidents. As discussed in Section 3.1 above, in the August 22, 2001, RAI response, the licensee stated that the reactor building purge system will be in operation such that any leakage through any penetration is from the environment into the reactor building since the purge systems maintain the reactor building at a negative pressure. The licensee also stated that should the purge system become inoperable, samples will be collected via installed portable air samplers located in the reactor building. Results of samples collected from the portable air samplers, in conjunction with conservative estimates of air flow out of the building, would be used to determine effluent releases. The licensee has committed to quantify these releases in accordance with its effluent control procedure 6610-ADM-4250.07, “Non-Routine Effluent Releases.” The licensee, therefore, complies with the requirements of GDC 64, and the proposed changes to TS Section 3.8.7 are acceptable.

3.3 Addition of TS Section 3.8.11

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS within containment, except during latching and unlatching of CONTROL ROD drive shafts, requires a minimum water level of 23 feet above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool to satisfy the FHA analysis assumptions. The addition of TS Section 3.8.11 reads as follows:

> “During the handling of irradiated fuel in the Reactor Building[,] at least 23 feet of water shall be maintained above the level of the reactor pressure vessel flange. If the water level is less than 23 feet above the reactor pressure vessel flange, place the fuel assembly(s) being handled into a safe position, then cease fuel handling until the water level has been restored to 23 feet or greater above the reactor pressure vessel flange.”

The licensee in its January 23, 2001, letter did not provide an associated surveillance requirement to ensure the minimum water level is maintained during refueling operations. In the August 22, 2001, RAI response, the licensee committed to revising its Surveillance Procedure 1301-1, “Shift and Daily Checks,” to require that during each shift, when moving irradiated fuel, that the fuel transfer canal level is verified to be greater than 23 feet.

The licensee’s commitment to check the water level by procedure does not provide adequate assurance that the proper water level will be maintained in the refuel canal. Licensee controlled procedures may be changed without NRC approval and would not have the same level of enforceability as a TS surveillance requirement. In a letter dated September 17, 2001, in response to NRC staff concerns, the licensee agreed to place the requirement to verify the minimum water level of 23 feet above the top of the reactor vessel flange in its Long Range Planning Program (LRPP) as a Category A schedule item. Per TMI-1 License Condition 2.c(9), LRPP Category A schedule items shall not be changed without prior approval from the NRC, and would have the same level of enforceability as a TS surveillance requirement.

The commitment to place the requirement to verify that the refuel canal water level is greater than 23 feet in the LRPP, as provided in the TMI-1 License Condition 2.c(9), is acceptable. The licensee may choose to relocate that surveillance requirement to the TSs during the next operating cycle. Based on the above, the licensee's proposed addition of TS Section 3.8.11 is acceptable.

3.4 Revised TS Section 4.4.1.3

The current TS Section 4.4.1.3 requires that operability of personnel and emergency air lock door interlocks and associated control room annunciator circuits be determined at least once per 6 months. In addition, the TS states that if the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable. The licensee is revising TS Section 4.4.1.3 to provide an exception to this requirement as specified in TS Section 3.8.6 (see Section 3.1 above).

The NRC staff finds the revision to TS Section 4.4.1.3 acceptable, in accordance with the findings of Section 3.1 above for the revision of TS Section 3.8.6.

3.5 Revised TS Bases Section 3.8

The NRC staff has reviewed the supporting bases revisions for the proposed TS changes, and agrees with the licensee that they are consistent with the proposed changes.

3.6 Confirmatory Dose Calculations

The requirements on containment air lock doors and other penetration closures ensure that a release of fission products from the containment resulting from an FHA will be restricted from release to the environment. During core alterations or movement of fuel assemblies within containment, the most limiting radiological consequences from a design-basis accident consideration result from a fuel-handling accident.

The licensee performed and submitted a radiological consequence analysis resulting from an FHA within the containment with the air lock doors and other penetrations open using the AST. The licensee concluded that the release of fission products, subsequent to an FHA, will result in doses that are well within the doses specified in 10 CFR 50.67 for the exclusion area boundary (EAB), low population zone (LPZ), and for the control room operators.

The licensee stated that it used the following assumptions in its analysis:

(1) assumption of one whole fuel assembly with the highest radial peaking factor is damaged releasing its fission products in the fuel gap into the reactor cavity water,

(2) assumption of a fission product decay period of 72 hours (time period from the reactor shutdown to the first fuel movement),

(3) assumption of an overall decontamination factor of 200 for the iodine isotopes in the reactor cavity water with minimum pool water depth of 23 feet above the top of the core, and

(4) conservative assumption of 1,000 cubic feet per minute (cfm) unfiltered air inleakage into the control room (based on trace gas testing).

The NRC staff has reviewed the licensee's analysis and finds that the major parameters and calculational methods used for the radiological consequence analysis are consistent with those provided in the Standard Review Plan, Section 15.0.1, "Radiological Consequence Analyses Using Alternative source Terms," and RG 1.183. In August 2000, the licensee performed a trace gas test to verify the unfiltered air inleakage rate into the control room following a design-basis accident. The testing was performed in accordance with ASTM E741-93 with the ventilation system in the emergency lineup configuration. The test results indicated that unfiltered inleakage flow rates were 233 $\pm$ 129 cfm for the "A" train and 189 $\pm$ 103 cfm for the "B" train. The licensee's 1,000 cfm inleakage rate used in dose calculation conservatively bounds measured unfiltered inleakage rate into the control room.

To verify the licensee's analyses, the NRC staff performed a confirmatory radiological consequence calculation. The resulting radiological consequences calculated by the NRC staff are shown in the attached Table 1, and the major parameters and assumptions used by the NRC staff are listed in the attached Table 2. The radiological consequences calculated by the NRC staff for the EAB, LPZ, and control room operators are consistent with those calculated by the licensee. Both, the radiological consequences calculated by the licensee and by the NRC staff are within the radiation doses set forth in 10 CFR 50.67.

On the basis of this evaluation, the NRC staff has determined that the radiological consequences analyzed and submitted by the licensee are acceptable. The implementation of an alternative source term in accordance with 10 CFR 50.67 and RG 1.183 in re-analyzing the FHA for TMI-1 is acceptable. The inclusion of the surveillance requirement to verify that the refuel canal water level is greater than 23 feet in the LRPP, as provided in License Condition 2.c(9), is acceptable. Additionally, the containment closure time of 45 minutes following a FHA further minimizes any potential offsite release associated with an FHA. Based on the above discussions, the NRC staff finds the licensee's proposed changes to the TSs requested in the January 23, 2001, application for amendment, as supplemented by letters dated August 22 and September 17, 2001, to be acceptable.

4.0 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.

5.0 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 and a surveillance requirement. 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 (66 FR 31702). 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.

6.0 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.

Principal Contributors: J. Lee
G. Hatchett

Date: October 2, 2001

TABLE 1

Radiological Consequences
for
Fuel Handling Accident
(rem as Total Effective Dose Equivalent)

	Calculated Doses	Acceptance Criteria
Exclusion Area Boundary	4.2	6.3[1]
Low Population Zone	0.7	6.3[1]
Control Room	0.5	5.0[2]

(1) Standard Review Plan 15.0.1
(2) 10 CFR 50.67

Table 2
Parameters and Assumptions Used in

Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor Power	2619 MWt
Radial peaking factor	1.7
Fission product decay period	72 hours
Number of fuel rods damaged	208
Number of fuel assemblies in core	177
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5%
Kr-85	10%
I-131	8%
Iodine except I-131	5%
Fuel Pool Decontamination Factors	
Iodine	200
Noble gases	1
Dose Conversion Factors	FGR 11 and 12
Control Room	
Volume	2.5E+5 ft^3
Unfiltered infiltration	1,000 cfm
Unfiltered outside air intake	
0 to 30 minutes	4,000 cfm
Filtered outside air intake	
30 minutes to 2 hours	8,000 cfm
Filtered recirculation flow	2.8E+4 cfm
Charcoal adsorber iodine removal efficiency	75%
Particulate filter removal efficiency	99%
Control room isolation time	30 minutes
Duration of accident	2 hours
Duration of fission product release	2 hours

Table 2 Continued

Atmospheric relative concentrations
(sec/m^3)

Exclusion Area Boundary	
0 to 2 hours	8.0E-4
Low Population Zone	
0 to 2 hours	1.4E-4
2 to 8 hours	6.0E-5
8 to 24 hours	3.9E-5
24 to 96 hours	1.6E-5
96 to 720 hours	4.0E-6
Control Room	
0 to 2 hours	3.40E-4
2 to 8 hours	2.25E-4
8 to 24 hours	1.02E-4
24 to 96 hours	7.16E-5
96 to 720 hours	4.99E-5

Three Mile Island Nuclear Station, Unit No. 1

cc:

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