September 19, 2001

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

SUBJECT: THREE MILE ISLAND NUCLEAR STATION UNIT-1 - ISSUANCE OF AMENDMENT RE: ENGINEERED SAFEGUARDS FEATURE (ESF) SYSTEMS LEAKAGE OUTSIDE CONTAINMENT (TAC NO. MB1074)

Dear Mr. Warner:

The Commission has issued the enclosed Amendment No. 235 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 29, 2001, as supplemented July 6, 2001.

The amendment removes the note from TMI-1 Technical Specification 4.5.4.1 which restricts the applicability of the specified Engineered Safeguards Feature Systems leakage rate limit of 15 gallons per hour to the current operating cycle (Cycle 13). The amendment also approves full scope implementation of an alternate source term for TMI-1 in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

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. 235 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

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Accession No. ML012480262 *SE input recieved. No changes made.

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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. 235
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 29, 2001, as supplemented July 6, 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. 235, 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 and shall be implemented within 30 days of issuance. The licensee's commitment to install a modified orifice in each Reactor Building Spray System pump discharge line to ensure spray flow is greater than 800 gallons per minute is consistent with accident analysis assumptions and shall be completed prior to startup of Cycle 14.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Peter Tam, 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: September 19, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 235

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
4-45	4-45

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235 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 29, 2001, and supplemented by letter dated July 6, 2001, Amergen Energy Company (the licensee) requested a license amendment for Three Mile Island Nuclear Station, Unit 1 (TMI-1). This license amendment request proposed to revise TMI-1 Technical Specification (TS) Section 4.5.4.1 to remove the existing note that restricts the applicability of the specified engineered safeguards features (ESF) system leakage rate limit of 15 gallons per hour (gph) to the current operating cycle (Cycle 13) and establish this leakage rate value as the permanent TS limit. The Nuclear Regulatory Commission (NRC) staff imposed this restriction in License Amendment No. 215, issued on August 24, 1999. The July 6, 2001, letter 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. A camera-ready copy of the TS page was provided by letter dated September 10, 2001.

2.0 BACKGROUND

The licensee previously requested, in License Amendment Request (LAR) No. 274, dated February 2, 1999, to increase the maximum allowable ESF system leakage to 15 gph from 0.6 gph following the postulated loss-of-coolant accident (LOCA). The NRC staff's safety evaluation (SE) of LAR No. 274 in support of Amendment No. 215 found that the resulting thyroid dose to the control room operator calculated by both the licensee and the staff exceeded the dose acceptance criterion specified in the Standard Review Plan (SRP). The SRP interpretation is that the appropriate acceptance criterion for thyroid dose to comply with General Design Criterion (GDC) 19 is 30 Rem. The SE supporting License Amendment No. 215 concluded as follows:

> The staff is closely working with the Nuclear Energy Institute Control Room Habitability Task Force to resolve generically all control room habitability related issues. Therefore, because the thyroid dose is not significantly exceeded (the relevant dose limit) and we expect to resolve control room habitability issues in the near future, we conclude that the amendment request 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.

The generic control room habitability issues referred to in Amendment No. 215 have not yet been resolved. Meanwhile, the licensee performed a trace gas test and verified the unfiltered air inleakage rate assumed in its control room operator dose calculation. The verification of the unfiltered air inleakage rate into the control room is a major factor in the resolution of control room habitability related issues. The NRC staff's acceptance of the licensee's unfiltered air inleakage rate into the control room here does not preclude any future generic regulatory actions that may result from forthcoming resolution of the remaining generic control room habitability issues.

In the January 29, 2001, license amendment request, the licensee also proposed full implementation of the alternative source term (AST) and provided revised TMI-1 offsite and control room operator doses using the proposed AST. The full implementation of the AST would replace the current accident source term used in the design-basis radiological consequence analyses with an AST pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident Source Term." As part of the full implementation of the AST, the total effective dose equivalent (TEDE) acceptance criteria in 10 CFR 50.67 replace the previous whole body and thyroid dose guidelines provided in 10 CFR Part 100 and GDC 19. Applicable portions of the NRC staff's SE in support of TMI-1 License Amendment No. 215 are repeated here since the licensee's request to increase the maximum allowable ESF system leakage in the TMI-1 TSs is the same in both license amendment requests.

3.0 EVALUATION

The licensee proposed full implementation of the AST pursuant to 10 CFR 50.67, and Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" for assessing the radiological consequence of the postulated LOCA. In its previous application (LAR No. 274), the licensee's radiological consequence analyses were based on an instantaneous release of fission products from the reactor core into the primary containment following the postulated LOCA as described in the Technical Information Document (TID)-14844 source term and the dose acceptance criteria specified in 10 CFR Part 100. Accordingly, the licensee also requested to change the TMI-1 TS Section 4.5.4 Bases to identify 10 CFR 50.67 as the dose acceptance criteria in lieu of 10 CFR Part 100.

To demonstrate the adequacy of the TMI-1 ESF systems designed to mitigate the radiological consequences of the design-basis accidents (DBAs) with the increased ESF system leak rate of 15 gph and without relying upon the auxiliary and fuel handling building ventilation system (AFHBVS) for removal of fission products, the licensee reevaluated the offsite and control room radiological consequences resulting from the most limiting LOCA using the AST. The licensee included the results of these offsite and control room dose calculations in the January 29 and July 6, 2001, submittals.

In the submittals, the licensee concluded that the existing ESF systems at TMI-1 with the increased ESF leak rate and without relying upon the AFHBVS for fission product removal still provide assurance that the radiological consequences at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room resulting from the postulated LOCA will be within the dose criteria specified in 10 CFR 50.67. The licensee calculated the radiological consequences for the following four potential fission product release pathways after the postulated LOCA:

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

The NRC staff has reviewed the licensee's radiological consequence analyses and finds that the methods and the major assumptions and parameters used are consistent with those provided in RG 1.183. Although the NRC staff performed independent dose calculations to confirm the licensee's results, the NRC staff's acceptance is based on the licensee's analyses. The licensee used the NRC computer code RADTRAD, Version 3.02, in calculating the radiological consequence doses. The results of the licensee's radiological consequence calculations are given in Table 1. The major parameters and assumptions used by the staff and the licensee in the radiological consequence calculations are listed in Tables 2 and 3.

3.1 Containment Leak Pathway

The licensee evaluated the radiological consequences resulting from containment leakage following a postulated design-basis LOCA. The licensee 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 in accordance with the guideline provided in RG 1.183. The source term in the containment is assumed to mix homogeneously throughout the free air volume of the containment.

The containment spray will dominate the removal of aerosol fission products in the containment atmosphere. In its July 6, 2001, submittal, the licensee revised the description in its January 29, 2001, submittal, of the reactor building spray operation as taking suction from the reactor building sump after the initial depletion of the BWST. In the revised reactor building spray operation, two spray pumps operate at 800 gpm, each taking borated water from the BWST after the 75-second startup time for 28.19 minutes until the BWST water is depleted. The NRC staff agrees with the licensee's assumption that the two-spray-pump and one-cooling-fan operation has the most conservative radiological consequences when considering the single failure criterion prior to the depletion of the BWST. Two-pump operation is more conservative than one-pump operation since it depletes borated water in the BWST more quickly and starts spray water recirculation earlier. Leakage from the ESF components is assumed to start as soon as spray water recirculation commences (see Section 3.2 below).

After the BWST is depleted, the spray pumps are realigned to take suction from the containment sump and recirculate sump water to continue spray operation. The licensee assumed one spray pump will be operational with a minimum flow rate of 800 gpm for 4 hours. The licensee committed in its July 6, 2001, letter to install a modified orifice, prior to Cycle 14 startup, in each reactor building spray pump discharge line to ensure that spray flow is greater than 800 gpm at all times throughout the design-basis accident. One-pump operation is more conservative since lower spray water flow reduces aerosol removal rate.

No fission product removal credit was taken for the FHBVS filters. The licensee assumed a mixing rate of approximately 2 unsprayed volumes per hour (25,000 cfm) between the sprayed and unsprayed regions of the containment atmosphere based on the operating capacity of one-out-of-three containment emergency cooling fans. The licensee also assumed aerosol removal in the unsprayed area of the containment by natural deposition, using the model provided in the RADTRAD code with a 10th percentile uncertainty distribution. The NRC staff finds these assumptions to be acceptable. The NRC staff finds that the radiological consequence contribution from this release pathway is the major source (approximately 89 percent) of the overall EAB dose resulting from the postulated LOCA (see Table 1).

3.2 Post-LOCA Leakage From Engineered Safeguards Features Systems

With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core. The total maximum allowable leakage into the auxiliary building from the applicable portions of the ESF system components is currently specified in TMI-1 TS Section 4.5.4.1 as 15 gph. A note further specifies that this leak rate limit is only applicable to operating Cycle 13. The licensee requested to remove this note and establish this limit as the permanent TS limit.

The licensee assumed that ESF system component leakage begins at 28.44 minutes into the accident (75-second spray pump startup time plus 28.19-minute BWST depletion time) at a leakage rate of 30 gph (two times the amount of the TS limit) for the entire duration of the accident (30 days), consistent with the guideline provided in RG 1.183. The staff accepted the licensee's assumption that 5 percent of the iodine in the ESF leakage water becomes airborne for the first 24 hours and 2 percent after 24 hours until the end of the 30-day accident period based on the containment sump water pH and the initial sump water temperature profiles. The airborne iodine is assumed to be released immediately to the environment. Radioiodine that is postulated to be available for release to the environment is assumed to be 97 percent in elemental iodine form and 3 percent in organic iodine form, consistent with RG 1.183. The staff finds that the radiological consequence associated with this release pathway is the major source (greater than 85 percent) of the overall control room operator dose resulting from the postulated LOCA (see Table 1).

3.3 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, decreasing to 102 gph until 24 hours, and then decreasing 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 NRC staff accepted the proposed leakage values. All of the ESF system leakage reaching the BWST is assumed to be in the liquid form. The licensee assumed an iodine partition factor of 10. No credit is taken for iodine plateout in the BWST. Also, as in the case of the ESF leakage into the auxiliary building, the licensee assumed that the radioiodine that is postulated to be available for release from the BWST to the environment is 97 percent in elemental iodine form and 3 percent in organic iodine form. The NRC staff finds that the radiological consequence contribution from this release pathway is a small fraction (less than 1 percent) of the overall offsite and control room operator doses (see Table 1).

3.4 Post-LOCA Containment Purge

Primary containment purging is allowed in the TMI-1 TSs during reactor power operation to reduce airborne radioactivity in order to facilitate containment entry. The licensee analyzed the release via this pathway assuming all of the radionuclides inventory in the reactor primary coolant system is released to the containment at the initiation of the postulated LOCA. The purge system was conservatively assumed to be isolated within 1 minute into the accident, with the reactor operating at the primary coolant iodine concentration of 1 microcurie per gram of dose equivalent iodine-131 and 1 percent defective fuel. The NRC staff finds these assumptions to be conservative and, therefore, acceptable. The staff further finds that the radiological consequence contribution from this release pathway is a small fraction (less than 1 percent) of the overall offsite and control room operator doses (see Table 1).

3.5 Control Room Habitability

The requirements for the protection of the control room operators under postulated accident conditions are specified in 10 CFR 50.67. The licensee has proposed to meet these requirements by incorporating shielding and emergency ventilation systems in the control room. Upon receipt of an ESF signal, the control room is automatically isolated and the normally operating ventilation fans are tripped. The control room emergency ventilation system is assumed to be manually initiated 30 minutes after the postulated LOCA. The licensee assumed a total of 4,000 cfm (cubic feet per minute) unfiltered air inleakage into the control room during the initial 30 minutes. After 30 minutes, the licensee assumed 1,000 cfm unfiltered air inleakage into the control room with 8,000 cfm filtered outside air intake to pressurize the control room. This places the control building emergency ventilation system in a recirculating mode with 28,000 cfm of control room air being circulated through redundant high-efficiency particulate air filters (99 percent removal efficiency) and charcoal adsorber units (90 percent iodine removal efficiency).

In August 2000, the licensee performed trace gas testing to verify the unfiltered air inleakage rate into the control room following a DBA. The testing was performed in accordance with ASTM E741-93 with the ventilation system in the emergency lineup configuration. The test results indicated that the unfiltered inleakage flow rates were 233 $\pm$ 129 cfm for the A train and 189 $\pm$ 103 cfm for the B train. The 1,000 cfm inleakage rate used in the dose calculation conservatively bounds the measured unfiltered inleakage rate into the control room. The results of the licensee's control room radiological consequence calculations are given in Table 1. The major parameters and assumptions used by the NRC staff in its confirmatory dose calculation and by the licensee in its dose calculation are listed in Tables 2 and 3. The radiological consequences to the control room operator calculated by the licensee and the NRC staff are within the dose criterion specified in 10 CFR 50.67.

3.6 Atmospheric Relative Concentrations at the Exclusion Area Boundary, in the Low Population Zone and in the Control Room

By letter dated October 15, 1998, as supplemented by a letter dated February 3, 1999, the licensee submitted a proposal to amend the atmospheric dispersion (χ/Q) values for the EAB and the LPZ for the radiological consequence assessments resulting from the postulated design basis accidents. The NRC staff reviewed the licensee's analysis and concluded that the proposed revision of the χ/Q values was acceptable in License Amendment No. 210, dated April 15, 1999. These revised χ/Q values, which are used in this evaluation, are listed in Table 3.

The licensee calculated the 95th percentile χ/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 the receptor increases, the calculational results of these two methodologies converge. Following discussions with the NRC 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 NRC 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 χ/Q values calculated making this assumption were about 75 percent higher than those calculated assuming the diffuse source release option. The χ/Q values calculated by the NRC 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, effluent could be released from more than one location (e.g., leaking from two penetrations) or over a small area.

The licensee has proposed using χ/Q values for a release from the BWST to the control room via the control room ventilation exhaust instead of the yard intake. The BWST is closer to the ventilation exhaust than to the yard intake, although in a different wind direction. The NRC staff confirmed that this assumption is more conservative than assuming intake at the yard intake. The licensee calculated χ/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 NRC staff performed confirmatory calculations. The NRC staff finds the χ/Q values calculated by the licensee to be acceptable for use in the radiological assessment. However, the NRC staff may reassess the control room χ/Q values proposed by the licensee for future license amendment requests, if needed, based on the forthcoming resolution of control room habitability issues and upon issuance of a new regulatory guide on control room meteorology in support of design-basis control room radiological habitability assessment.

The NRC staff has reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequences resulting from the postulated LOCA. Although the NRC staff performed independent calculations to confirm the licensee's results, the NRC staff's acceptance is based on the licensee's analyses. The doses calculated by the licensee are listed in Table 1. The major parameters and assumptions used by the licensee and the NRC staff are listed in Tables 2 and 3. The NRC staff's analysis confirmed the licensee's conclusion

that the radiological consequences would not exceed the dose criteria specified in 10 CFR 50.67 for the EAB and LPZ and for the control room operator.

Therefore, the NRC staff has determined that the license amendment requested by the licensee to delete the existing note in TMI-1 TS Section 4.5.4.1 that limits the applicability of the specified ESF system leakage limit of 15 gph to the current operating Cycle 13 and to establish this value as the permanent TS limit is acceptable. The bases for the staff's acceptance are: (1) that the resulting radiological consequences from the postulated LOCA are within dose criteria specified in 10 CFR 50.67, (2) that the methodology used by the licensee for dose calculations is consistent with the guidelines provided in RG 1.183, and (3) the licensee's commitment in its July 6, 2001, letter, to install a modified orifice, prior to Cycle 14 startup, in each reactor building spray system pump discharge line to ensure that spray flow is greater than 800 gpm at all times throughout the design-basis accident.

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. 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 31703). 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.

Attachments: (1) Table 1, Radiological Consequences
(2) Table 2, Assumptions Used in Computing Radiological Consequences
(3) Table 3, Atmospheric Dispersion Factors, (χ/Q values)

Principal Contributor: J. Lee

Date: September 19, 2001

TABLE 1

Radiological Consequences[1]
(rem as TEDE (total effective dose equivalent))

Release Pathways	EAB	LPZ	Control Room
Containment leak	21.7	5.76	0.64
ECCS leak	2.62	1.91	4.06
BWST vent	0.031	0.045	0.037
Containment purge	0.026	0.005	0.004
Control Room filter shine	N/A	N/A	0.012
TOTAL	24.4	7.72	4.75
Dose Criteria[2]	25	25	5

[1] Rounded to 3 significant figures
[2] From 10CFR 50.67

Table 2
Assumptions Used in Computing Radiological Consequences

Parameter	Value	
Power level, MWt	2619	
Fraction of core inventory released	Regulatory Guide 1.183 (Table 2)	
Iodine chemical forms	Regulatory Guide 1.183 (Section 3.5)	
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	1.23E+6	
Unsprayed volume	9.29E+5	
Containment spray		
Flow rate, gpm		
75 seconds to 28.19 minutes	1600	
28.19 minutes to 4 hours	800	
Iodine removal rate by spray, $hour^{-1}$	Aerosol	Elemental
75 seconds to 28.19 minutes	3.88	9.46
28.19 minutes to 2.92 hours	1.94	4.73
2.92 to 3.82 hours	1.94	0
3.82 to 4 hours	0.194	0
4 to 720 hours	0	0
Containment atmosphere mixing rate, cfm		
Sprayed to unsprayed	2.5E+4	
Unsprayed to sprayed	2.5E+4	
Sump water volume, ft^3	5.45E+4	
Sump water recirculating startup time, minutes	28.19	
ECCS leak rate, gallons per hour	30	
Iodine partition factor for ECCS water leaked into containment, %		
0 to 24 hours	5	
24 to 720 hours	2	
Iodine chemical form in ECCS leakage released to environment, %		
Elemental	97	
Organic	3	

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

BWST volume, ft^3	4.07E+4
Sump water leakage to BWST, gpm	
0 to 5 hours	3
5 to 24 hours	1.7
24 to 720 hours	1.6
Iodine partition factor for ECCS water leaked into BWST, %	
0 to 720 hours	10
Iodine chemical form in ECCS leakage released to BWST, %	
Elemental	97
Organic	3

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

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

Control Room χ/Q Values (sec/m^3)

Containment Leakage

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

ESF Leakage

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

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

Three Mile Island Nuclear Station, Unit No. 1

cc:

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