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Three Mile Island Unit 1
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An Exelon/British Energy Company

10 CFR 50.90
10 CFR 50.67

January 23, 2001

5928-00-20299

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Dear Sir or Madam:

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
LICENSE AMENDMENT REQUEST NO. 249
CONTAINMENT INTEGRITY DURING REFUELING OPERATIONS

In accordance with 10 CFR 50.4(b)(1), enclosed is License Amendment Request No. 249.

The purpose of this License Amendment Request is to revise the TMI Unit 1 Technical Specification requirements for containment integrity associated with the personnel and emergency air locks and other penetrations during fuel movement and refueling operations. This change will improve the reliability of the personnel and emergency air lock doors, improve personnel safety, and allow more efficient plant refueling outages. The basis for this change is a reanalysis of the limiting design basis Fuel Handling Accident Inside Containment using an Alternative Source Term in accordance with 10 CFR 50.67 and Regulatory Guide 1.183. Therefore, the proposed change also requests NRC approval of selective implementation of Alternative Source Term methodology for the TMI Unit 1 Fuel Handling Accident Inside Containment design basis accident analysis. The TMI Unit 1 Updated Final Safety Analysis Report Section 14.2.2.1, Fuel Handling Accident, design basis accident analysis description will be updated to reflect the enclosed reanalysis upon NRC approval. A full scope application of an Alternative Source Term is being submitted separately via TMI Unit 1 License Amendment Request No. 290.

A001

Using the standards in the 10 CFR 50.92, AmerGen Energy Company, LLC (AmerGen) has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), a copy of this License Amendment Request is provided to the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection, as well as the chief executives of the township and county in which the facility is located.

Approval of this license amendment is requested by August 1, 2001, to support the detailed planning and scheduling activities for the TMI Unit 1 14R refueling outage scheduled for September 2001.

If any additional information is needed, please contact David J. Distel at (610) 765-5517.

Sincerely yours,

Mark E. Warner
Vice President, TMI Unit 1

MEW/djd

Enclosures: (1) TMI Unit 1 License Amendment Request No. 249
Safety Evaluation and No Significant Hazards Consideration
(2) Affected TMI Unit 1 Technical Specification Pages

cc: USNRC, Regional Administrator, Region I
USNRC, TMI Unit 1 Senior Project Manager
USNRC, TMI Senior Resident Inspector
Director, Bureau of Radiation Protection - PA Department of Environmental Resources
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township
File No. 00109

AMERGEN ENERGY COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
License Amendment Request No. 249

COMMONWEALTH OF PENNSYLVANIA)
) **SS:**
COUNTY OF DAUPHIN)

This License Amendment Request is submitted in support of Licensee's request to change the Technical Specifications for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed marked up pages for the TMI Unit 1 Technical Specifications are also included. All statements contained in this submittal have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

AmerGen Energy Company, LLC

BY: [signature]
Vice President, TMI Unit 1

Sworn and Subscribed to before me

this 23rd day of January 2001.

[signature]
Notary Public

Notarial Seal
Suzanne C. Miklosik, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Nov. 22, 2003
Member, Pennsylvania Association of Notaries

ENCLOSURE 1

TMI Unit 1 License Amendment Request No. 249

Safety Evaluation and No Significant Hazards Consideration

I. License Amendment Request No. 249

AmerGen Energy Company, LLC (AmerGen) requests that the following changed replacement pages be inserted into the existing Technical Specification:

Revised Technical Specification pages: 3-44, 3-45 and 4-29

Marked up pages showing the requested changes are provided in Enclosure 2.

II. Reason for Change

The purpose of this License Amendment Request is to revise the TMI Unit 1 Technical Specification requirements for containment integrity during fuel movement and refueling operations, and to request NRC approval for a change to the TMI Unit 1 licensing basis to incorporate Alternative Source Term methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," July 2000, for the design basis Fuel Handling Accident Inside Containment. The reanalysis of the Fuel Handling Accident Inside Containment using an Alternative Source Term supports the proposed Technical Specification change.

Technical Specification Sections 3.8.6 and 3.8.7 are revised to allow the personnel and emergency air lock doors and other containment penetrations to remain open during fuel movement and refueling operations. The proposed Technical Specification would require that during fuel movement and refueling operations at least one door in each of the personnel and emergency air locks is capable of being closed and that each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be capable of being closed by an isolation valve, blind flange or manual valve. A footnote is added to Technical Specification Sections 3.8.6 and 3.8.7 to identify the requirement for administrative controls associated with closure of the personnel and emergency air lock doors and other mechanical penetrations. The existing Technical Specification requirements for containment purge and exhaust isolation valve closure are not affected by this proposed change. This change will improve the reliability of the personnel and emergency air lock doors, improve personnel safety, and allow more efficient plant refueling outages.

Technical Specification Section 3.8.7 is also revised to provide equivalent isolation methods for other penetrations consistent with B&W Owners Group Standard Technical Specifications, Section 3.9.3.c.1, NUREG-1430, April 1995.

Technical Specification Section 3.8.11 is added 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.

Technical Specification Section 3.8 Bases is 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. Technical Specification 3.8 Bases is also revised to provide the description of equivalent isolation methods for other penetrations consistent with B&W Owners Group Standard Technical Specifications, Section 3.9.3, NUREG-1430, April 1995.

Technical Specification Section 4.4.1.3 surveillance requirement is also revised to identify the exception allowed by Technical Specification Section 3.8.6 under which both doors of the personnel and emergency air locks can be open.

III. Safety Evaluation Justifying Change

The TMI Unit 1 containment is equipped with two (2) personnel and emergency air locks. TMI Unit 1 Technical Specification Sections 3.8.6 and 3.8.7 require a minimum of one door in each of the containment personnel and emergency air locks, as well as other containment penetrations, be closed during fuel loading and refueling operations within the reactor containment building. The purpose of this requirement is to mitigate the consequences of a fuel handling accident inside containment. Current fuel handling accident analysis assumptions are reflected in Section 14.2.2.1 of the TMI Unit 1 Updated Final Safety Analysis Report (UFSAR). AmerGen has performed a new analysis of the fuel handling accident utilizing Alternative Source Term methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183, which assumes that the containment personnel and emergency air lock doors and other penetrations, including the equipment hatch, are open at the time of the accident.

During a refueling outage, additional work inside the containment continues during fuel movement and core alterations, requiring frequent cycling of the containment personnel and emergency air lock doors in order to enter and exit containment. Repeated cycling and heavy use of the containment personnel and emergency air lock doors necessitates substantial maintenance on the containment personnel and emergency air lock doors and components. Thus, leaving the personnel and emergency air lock doors open should increase the reliability of the air locks. Other mechanical penetrations are utilized to provide access for equipment and communications cables during a refueling outage. The proposed change to leave the personnel and emergency air lock doors and other mechanical penetrations open during the entire refueling outage would increase personnel and emergency air lock door reliability, provide greater efficiency in the movement of personnel and equipment, and result in decreased outage critical path time. This results in significant cost savings over the life of the plant.

Should a fuel handling accident occur, it would take a number of cycles of the personnel and emergency air lock doors to evacuate personnel from within containment. This would delay the evacuation of personnel from the containment. The containment could be evacuated more expeditiously with the personnel and emergency air lock doors open, thus, enhancing personnel safety. This would reduce dose to the workers in the event of an accident while maintaining acceptable doses to the public.

During the movement of irradiated fuel, administrative controls will be in place to assure closure of at least one door in each air lock, as well as other open containment penetrations, following a containment evacuation. These administrative control requirements are identified in the proposed change to Technical Specification Sections 3.8.6 and 3.8.7.

The NRC has approved a similar change for Arkansas Nuclear One, Unit 1 on September 20, 1996 (Amendment No. 184) and Unit 2 on September 28, 1995 (Amendment No. 166) for personnel and emergency air lock doors.

Revised Fuel Handling Accident Occurring in the Reactor Building

An evaluation of the postulated Fuel Handling Accident Inside Containment (FHAIC) at TMI Unit 1 has been performed utilizing Alternate Source Term methodology in accordance with 10 CFR 50.67 and following the guidance of Regulatory Guide 1.183. No credit is taken for containment integrity in terms of containment isolation, personnel and emergency air lock closure, or equipment hatch closure. No credit is taken for the Reactor Building Purge Exhaust System filtration prior to release to the environment.

The accident is assumed to happen after the reactor has been shut down for 72 hours. This is based on Technical Specification 3.8.10, which requires at least 72 hours between reactor shutdown and the removal of irradiated fuel. Radioactive decay of the core fission product inventory during this interval is taken into account based on the existing FHAIC analysis in the TMI Unit 1 Updated Final Safety Analysis Report (UFSAR) Section 14.2.2.1.

All the rods in one assembly (208 rods) are assumed to rupture as a result of the drop of one assembly into the reactor cavity. The damaged fuel is assumed to lie on top of the core in the reactor vessel with at least 23 feet of water above the top of the core. The assembly damaged is assumed to be the highest powered assembly in the core region to be discharged with a radial peaking factor of 1.7. These assumptions are identical with the existing TMI Unit 1 FHAIC analysis in UFSAR Section 14.2.2.1. In accordance with Regulatory Guide 1.183, Position 3.3 for non-LOCA design basis accidents (DBA's) in which fuel damage is assumed, the release from the fuel gap and the fuel pellet is assumed to occur instantaneously with the onset of fuel damage. In accordance with Regulatory Guide 1.183, Appendix B.1.3, the chemical form of radioiodine released from the fuel to the water is assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide; the CsI released from the fuel is assumed to completely dissociate in the water, and the iodine instantaneously re-evolves as elemental iodine due to the low pH of the water.

In accordance with Regulatory Guide 1.183, Appendix B.2, a decontamination factor of 200 is assumed for the elemental and organic iodine with a minimum depth of water above the damaged fuel of 23 feet or greater. This minimum water level is incorporated as a requirement for fuel loading and refueling operations in the proposed Technical Specification Section 3.8.11. In accordance with Regulatory Guide 1.183, Appendix B.3, the retention of noble gases in the water in the reactor cavity is negligible and a decontamination factor of 1 is assumed, and particulate radionuclides are assumed to be retained by the water in the reactor cavity.

In accordance with Regulatory Guide 1.183, Appendix B.5.3, activity released from the refueling cavity is uniformly distributed in the entire volume of the containment building and released to the environment over a two-hour time period such that 99.99 percent of the activity, released from the damaged fuel assembly is released to the environment. The post-FHAIC activity in the reactor building utilized in the RADTRAD Code is provided in attached Table 1 (AmerGen Calculation No. C-1101-900-E000-083).

Atmospheric dispersion coefficients (X/Q) for Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room operator doses are identical with the existing UFSAR accident analysis values previously approved by NRC in Amendment No. 210, dated April 15, 1999 and Amendment No. 215, dated August 24, 1999. The X/Q's for the control room were developed using the methodology of the NRC computer code ARCON96, as previously approved by NRC in TMI Unit 1 Amendment No. 215, dated August 24, 1999.

The design inputs and assumptions utilized in the EAB, LPZ, and Control Room Habitability analyses are listed in the attached Table 2 (AmerGen Calculation C-1101-900-E000-083). In accordance with Regulatory Guide 1.183, Position 5.1.4, these design inputs are compatible with the Alternate Source Term characteristics and TEDE dose criteria, and the assumptions are based on Regulatory Guide 1.183, Position 3, and Appendix B guidance.

The TMI Unit 1 control room emergency filtration system including the charcoal and HEPA filters are credited with 75 percent and 99 percent efficiency, respectively. These assumptions are bounded by the existing TMI Unit 1 Technical Specification, Section 3.15, requirements of 95% and 99.95% for the charcoal and HEPA filter efficiencies, respectively. The control room is assumed to be manually isolated and the emergency ventilation system to be manually initiated 30 minutes after the postulated accident occurs. This is conservative since the actions necessary to accomplish this are located within the Main Control Room on the H&V Panel. The control building envelope free air volume is 250,000 ft^3. Control room emergency ventilation system flow rates of 8,000 cfm maximum outside air intake and 28,000 cfm minimum recirculation flow are conservatively assumed based on system testing. A control building envelope unfiltered inleakage rate of 1000 cfm is also assumed which represents a factor of approximately three over the TMI Unit 1 tracer gas test results. TMI Unit 1 control building envelope tracer gas testing was performed in August 2000 to establish a measured unfiltered inleakage rate. This testing was performed in accordance with ASTM E741-93 with the ventilation system in the emergency lineup configuration. Unfiltered inleakage flow rates were determined to be 233 ± 129 scfm for the "A" ventilation train and 189 ± 103 scfm for the "B" ventilation train. Tracer gas methods also quantified the maximum outside air supply flowrate. Consistent with the existing TMI Unit 1 licensing basis, this testing also confirmed that all rooms inside the control building envelope were at a positive pressure relative to adjacent areas outside the envelope, and the main control room was maintained at a positive pressure of at least 0.1 inches w.g. with respect to adjacent areas of the control building envelope. These test results verify that the control building envelope is being adequately maintained, and that the proposed analysis conservatively bounds measured unfiltered inleakage into the control room. Additionally, TMI Unit 1 has performed system modifications to eliminate single active failure modes in the control building ventilation system dampers that contributed to unfiltered inleakage into the control building envelope. These modifications have improved system reliability and increased system leak tightness which have enhanced overall control room habitability performance. The tracer gas test included the ventilation equipment rooms in the control building envelope via temporary modifications. Permanent modifications will be completed prior to the start of the 14R outage to make these rooms a permanent part of the control building envelope.

The resulting dose consequences for the revised TMI Unit Fuel Handling Accident Inside Containment, using Alternate Source Term methodology in accordance with NRC Regulatory Guide 1.183 is tabulated below. These doses remain well within the allowable dose criteria as specified in Regulatory Guide 1.183 and 10CFR50.67.

	Fuel Handling Accident Inside Containment TEDE Dose (Rem)		
	Control Room	EAB	LPZ
Calculated Dose	6.55E-01	4.20E+00	7.35E-01
Allowable Dose	5.00E+00	6.30E+00	6.30E+00

In the event of a fuel handling accident, actual control room and offsite doses will be less because at least one door in each airlock, as well as other open penetrations, will be closed following an evacuation of containment. The analyses show that it is not necessary to have containment closure in order to show acceptable offsite or control room operator doses following a fuel handling accident.

The added provision for equivalent isolation methods for other penetrations is considered an administrative change consistent with B&W Owners Group Standard Technical Specifications, NUREG-1430, April 1995. This change enhances the existing TMI Unit 1 Technical Specifications by explicitly recognizing approved equivalent isolation methods.

IV. Environmental Consideration

AmerGen has determined that this change to the TMI Unit 1 Technical Specifications and Fuel Handling Accident Inside Containment dose analysis involves no significant hazards consideration, no significant change in the amount or type of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Although there is an increase in the amount of radioactivity released, this increase is not considered significant because the new consequences of the revised Fuel Handling Accident Inside Containment analysis remain well below the acceptance criteria specified in 10CFR50.67 and Regulatory Guide 1.183. As such, operation of TMI Unit 1 in accordance with the proposed change does not require an environmental assessment or an environmental impact statement.

V. No Significant Hazards Consideration

AmerGen has determined that this License Amendment Request poses no significant hazards considerations as defined by 10 CFR 50.92.

1. Will the operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response:

   The proposed change would allow the personnel and emergency air lock doors and other penetrations to remain open during fuel loading and refueling operations. These penetrations were previously closed during this time period in order to prevent the escape of radioactive material in the event of a fuel handling accident inside containment (FHA). These penetrations are not initiators of any accident. The probability of a FHA is unaffected by the position of these penetrations.

   The new FHA analysis utilizing an Alternative Source Term with an open containment demonstrates that the maximum doses are well within the acceptance criteria specified in 10CFR50.67 and Regulatory Guide 1.183. In the event of a fuel handling accident, actual control room and offsite doses will be less than the analyzed values because containment integrity will be restored following an evacuation of containment. As noted above, with the Alternate Source Term implementation, the acceptance criteria are also being revised. A direct comparison of the new Alternate Source Term dose consequences with the existing licensing basis FHA source term dose consequences is not practical due to the significant differences in methodology and assumptions. However, a comparison of the previous thyroid and whole body dose results for the postulated TMI Unit 1 FHA Inside Containment documented in the TMI Unit 1 UFSAR Chanter 14 with the new dose results expressed in terms of Total Effective Dose Equivalent (TEDE), using the guidance in Regulatory Guide 1.183 Footnote 7, indicates that the new doses are not significantly higher than the previous dose results. The revised Alternate Source Term calculated doses remain well within the allowable acceptance criteria.

Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will the operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed change does not involve the addition or modification of any plant equipment. Also, the proposed change would not alter the design or method of operation of the plant beyond the standard functional capabilities of the equipment. The proposed change involves a change to the Technical Specifications that would allow the personnel and emergency air lock doors and other penetrations to be open during fuel loading and refueling operations within the containment. Having these doors and penetrations open does not create the possibility of a new accident. Administrative provisions will be made to ensure the capability to close the containment in the event of a FHA inside containment.

Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will the operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response:

This proposed change has the potential for an increased postulated accident dose due to a FHA Inside Containment; however, the analysis demonstrates that the resultant doses are well within the appropriate acceptance criteria. The margin of safety, as defined by 10 CFR 50.67 and Regulatory Guide 1.183, has been maintained. The offsite and control room doses due to a FHA with an open containment have been evaluated with conservative assumptions, which ensure the calculation bounds the postulated accident dose. Closing at least one door in each of the personnel and emergency air locks following an evacuation of the containment and closure of other open penetrations would reduce the control room and offsite doses in the event of a FHA inside containment and provides additional margin to the calculated doses.

Therefore, the proposed change will not involve a significant reduction in a margin of safety.

VI. Implementation

It is requested that the amendment authorizing this change become effective upon issuance.

TABLE 1

Post-FHA Inside Containment Activity In Reactor Building - RADTRAD Code Nuclide Inventory File

Isotope	Core Initial Inventory (Ci) A	Radial Peaking Factor B	Number Of Fuel Assembly In Core C	Activity In Damaged Spent Fuel Assembly (Ci) D=A*B/C	DF E	Post-FHA Inside Containment For RADTRAD Code Nuclide Inventory File (Ci) F=D/E	(Ci/MWt) G=F/2619	(Ci/MWt) G*1
KR 85*	1.05E+06	1.7	177	1.00E+04	1.0	1.00E+04	3.833E+00	.7666E+01
KR 85M	2.33E+07	1.7	177	2.24E+05	1.0	2.24E+05	8.538E+01	.8538E+02
KR 87	4.60E+07	1.7	177	4.41E+05	1.0	4.41E+05	1.686E+02	.1686E+03
KR 88	6.48E+07	1.7	177	6.23E+05	1.0	6.23E+05	2.377E+02	.2377E+03
RB 86	1.64E+05	1.7	177	1.58E+03	200.0	7.89E+00	3.011E-03	.3011E-02
I131**	7.15E+07	1.7	177	6.87E+05	200.0	3.43E+03	1.311E+00	.2097E+01
I132	1.03E+08	1.7	177	9.92E+05	200.0	4.96E+03	1.894E+00	.1894E+01
I133	1.50E+08	1.7	177	1.44E+06	200.0	7.18E+03	2.741E+00	.2741E+01
I134	1.66E+08	1.7	177	1.60E+06	200.0	7.98E+03	3.048E+00	.3048E+01
I135	1.39E+08	1.7	177	1.34E+06	200.0	6.70E+03	2.556E+00	.2556E+01
XE-131M	7.17E+05	1.7	177	6.89E+03	1.0	6.89E+03	2.629E+00	.2629E+01
XE-133M	4.56E+06	1.7	177	4.38E+04	1.0	4.38E+04	1.672E+01	.1672E+02
XE133	1.50E+08	1.7	177	1.44E+06	1.0	1.44E+06	5.501E+02	.5501E+03
XE135	5.51E+07	1.7	177	5.29E+05	1.0	5.29E+05	2.021E+02	.2021E+03
XE135M	2.85E+07	1.7	177	2.74E+05	1.0	2.74E+05	1.045E+02	.1045E+03
CS134	1.71E+07	1.7	177	1.64E+05	200.0	8.22E+02	3.140E-01	.3140E+00
CS136	4.74E+06	1.7	177	4.55E+04	200.0	2.27E+02	8.684E-02	.8684E-01
CS137	1.15E+07	1.7	177	1.10E+05	200.0	5.51E+02	2.104E-01	.2104E+00

*KR-85 activity is multiplied by a factor 2(0.1/0.05) to account for additional fractional release.
**I-131 activity is multiplied by a factor 1.6(0.08/0.05) to account for additional fractional release.

TABLE 2

FHA Inside Containment With Airlocks, Hatch Doors, and Other Penetrations Opened Design Input Parameters

Design Input Parameter	Value Assigned
Source Term	
Power level	2,619 MWt (2.568 x 1.02 = 2,619)
Isotopic Core Inventory	See Below

Core Inventory					
Isotope	**Activity**	**Isotope**	**Activity**	**Isotope**	**Activity**
KR 85	1.05E+06	I132	1.03E+08	XE133	1.50E+08
KR 85M	2.33E+07	I133	1.50E+08	XE135	5.51E+07
KR 87	4.60E+07	I134	1.66E+08	XE135M	2.85E+07
KR 88	6.48E+07	I135	1.39E+08	CS134	1.71E+07
RB 86	1.64E+05	XE-131M	7.17E+05	CS136	4.74E+06
I131	7.15E+07	XE-133M	4.56E+06	CS137	1.15E+07

Radionuclide Release Factions Fraction of Fission Product Inventory in Gap	
Group	**Fraction**
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Radionuclide Composition	
Group	**Elements**
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Radial Peaking Factor	1.7
Damage Fuel Rods	208
Number of Fuel Assemblies	177
Irradiated Fuel Decay	72 hrs

Design Input Parameter	Value Assigned
Activity Transport in Reactor Building	
Water Depth	23 feet
Reactor Building Volume	2.16E+06 ft^3
Decontamination Factors (DFs)	
Elemental	500
Organic	1
Overall Effective Decontamination Factors (DFs)	
Elemental	200
Organic	1
Chemical Form of Iodine Released From Pool Water	
Elemental	57%
Organic	43%
DF of Noble Gas	1
Duration of Release (Hr)	2
Reactor Building Filter Efficiencies (Assumed In This Analysis)	
Species	**Efficiency (%)**
Aerosol	0
Elemental	0
Organic	0
Activity release rate (cfm)	165,780

TABLE 2 (Cont'd)

Design Input Parameter	Value Assigned
Control Room Model Parameters	
CR Volume	250,000 ft^3
CREV System Flow Rate	8,000 cfm
CR Min Recir Flow Rate	28,000 cfm
CR Unfiltered Inleakage	1,000 cfm
CR Charcoal Filter Efficiencies	75%
CR HEPA Filter Efficiency	99%

CR Occupancy Factors	
Time (Hr)	**%**
0 - 24	100
24 - 96	60
96 - 720	40
CR Breathing Rate (m^3/sec)	3.50E-04

CR Atmospheric Dispersion Factors (X/Qs)	
Time (Hr)	**(X/Q (sec/m^3)**
0 - 2	3.40E-04
2 - 8	2.25E-04
8 - 24	1.02E-04
24 - 96	7.16E-05
96 - 720	4.99E-05
Site Boundary Release Model Parameters	
EAB Atmospheric Dispersion Factor (X/Q) (sec/m^3)	8.0E-04

LPZ Atmospheric Dispersion Factors (X/Qs)	
Time (Hr)	**(X/Q (sec/m^3)**
0 - 2	1.4E-04
2 - 8	6.0E-05
8 - 24	3.9E-05
24 - 96	1.6E-05
96 - 720	4.0E-06

Breathing Rate (m^3/sec)	
Time (Hr)	**(m^3/sec)**
0 - 8	3.5E-04
8 - 24	1.8E-04
24 - 720	2.3E-04

ENCLOSURE 2

Affected TMI Unit 1 Technical Specification Pages

3.8 FUEL LOADING AND REFUELING

Applicability: Applies to fuel loading and refueling operations.

Objective: To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the Reactor Building refueling area shall be monitored by RM-G6 and RM-G7. Radiation levels in the spent fuel storage area shall be monitored by RM-G9. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one decay heat removal pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the Reactor Building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the Reactor Building at least one door in each of the personnel and emergency air locks shall be capable of being closed.* The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 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 equivalent, or capable of being closed,* or
2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

3.8.8 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

* INSERT A

Amendment No. ~~27, 198~~

INSERT A TO PAGE 3-44:

*Administrative controls shall ensure that appropriate personnel are aware that air lock doors and/or other penetrations are open, a specific individual(s) is designated and available to close the air lock doors and other penetrations as part of a required evacuation of containment. Any obstruction(s) (e.g., cables and hoses) that could prevent closure of an air lock door or other penetration will be capable of being quickly removed.

3.8.9 The reactor building purge system, including the radiation monitors which initiate purge isolation, shall be tested and verified to be operable no more than one week prior to refueling operations.

3.8.10 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

3.8.11 INSERT B

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the UFSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration. The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core (Reference 1). The boron concentration will be sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, however only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

INSERT C

The specification requiring testing Reactor Building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.10 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours (Reference 2).

REFERENCES

(1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"

(2) UFSAR, Section 14.2.2.1(2) - "FHA Inside Containment"

INSERT B TO PAGE 3-45:

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.

INSERT C TO PAGE 3-45:

Per Specification 3.8.6 and 3.8.7, the personnel and emergency air lock doors, and penetrations may be open during movement of irradiated fuel in the containment provided a minimum of one door in each of the air locks, and penetrations are capable of being closed in the event of a fuel handling accident, and the plant is in REFUELING SHUTDOWN or REFUELING OPERATION with at least 23 feet of water above the fuel seated within the reactor pressure vessel. The minimum water level specified is the basis for the accident analysis assumption of a decontamination factor of 200 for the release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel. Should a fuel handling accident occur inside containment, a minimum of one door in each personnel and emergency air lock, and the open penetrations will be closed following an evacuation of containment. Administrative controls will be in place to assure closure of at least one door in each air lock, as well as other open containment penetrations, following a containment evacuation.

Provisions for equivalent isolation methods in Technical Specification 3.8.7 include use of a material (e.g. temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for other containment penetrations during fuel movements.

4.4 REACTOR BUILDING

4.4.1 CONTAINMENT LEAKAGE TESTS

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Testing (ILRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program at test frequencies established in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.2 Local Leakage Rate Testing (LLRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program. LLRT shall be performed at a pressure not less than peak accident pressure P_{ac} with the exception that the airlock door seal tests shall normally be performed at 10 psig and the periodic containment airlock tests shall be performed at a pressure not less than P_{ac}. LLRT frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.3 Operability of the personnel and emergency air lock door interlocks and the associated control room annunciator circuits shall be determined at least once per six months. 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, except as provided in Technical Specification Section 3.8.6.

Bases (1)

The Reactor Building is designed to limit the leakage rate to 0.1 percent by weight of contained atmosphere in 24 hours at the design internal pressure of 55 psig with a coincident temperature of 281°F at accident conditions. The peak calculated Reactor Building pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig. The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}. Containment Isolation Valves are addressed in the UFSAR (Reference 2).