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10 CFR 50.90

July 29, 2005
5928-05-20176

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

Three Mile Island, Unit 1 (TMI Unit 1)
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Response To Request For Additional Information – Technical Specification Change Request No. 326: Elimination of Containment Equipment Hatch Closure During Refueling (TAC NO. MC4904)

This letter provides additional information in response to the NRC draft request for additional information received via NRC email, dated June 30, 2005, regarding TMI Unit 1 Technical Specification Change Request No. 326, submitted to NRC for review on October 20, 2004. The additional information is provided in Enclosure 1.

Revised regulatory commitments established by this submittal are identified in Enclosure 3. If any additional information is needed, please contact David J. Distel (610) 765-5517.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

7/29/05
Executed On

Pamela B. Cowan
Director - Licensing & Regulatory Affairs
AmerGen Energy Company, LLC

Enclosures:
1) Response to Request for Additional Information
2) TMI Unit 1 Calculation No. C-1101-900-E000-083, Rev. 2, "EAB, LPZ, and CR Doses Due to Fuel Handling Accident In Reactor Building With Hatch Doors Opened"
3) List of Commitments

cc: S. J. Collins, USNRC Administrator, Region I
P. S. Tam, USNRC Senior Project Manager, TMI Unit 1
D. M. Kern, USNRC Senior Resident Inspector, TMI Unit 1
File No. 04092

A001

ENCLOSURE 1

TMI UNIT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATION CHANGE REQUEST No. 326
ELIMINATION OF CONTAINMENT EQUIPMENT HATCH CLOSURE DURING REFUELING

1. **NRC Question**

What design-basis parameters, assumptions or methodologies were changed in the radiological design-basis accident analyses as a result of the proposed change? If there are many changes it would be helpful to compare and contrast them in a table. Also, please provide a justification for any changes.

Response

The TMI Unit 1 Fuel Handling Accident Inside Containment (FHAIC) radiological analysis supporting alternative source term implementation for the FHAIC was previously approved in TMI Unit 1 Amendment No. 236, dated October 2, 2001. This analysis is contained in AmerGen Calculation No. C-1101-900-E000-083, Rev. 1, dated January 19, 2001. This analysis was not revised to support the proposed change for the containment equipment hatch originally submitted in Technical Specification Change Request No. 326. However, this calculation has been subsequently revised to address additional concerns identified in the following NRC questions, including the revised bounding control room air intake X/Q value described in the TMI Unit 1 response to the NRC request for additional information submitted to NRC on June 30, 2005.

This revision to Calculation No. C-1101-900-E000-083 is provided in Enclosure 2 and supersedes the previous analysis described above. The parameters and assumptions used in the analysis are provided in Sections 5.2 through 5.5 of Calculation No. C-1101-900-E000-083, Rev. 2.

A summary of parameter changes from the previous analysis is provided below:

Parameter	Previous Analysis	Revised Analysis	Reason/Justification for Change
Fraction of fission product inventory in gap	Two times the iodine value was used. Cesium was also included for conservatism.	Two times the iodine and noble gas values were used without contribution due to cesium.	Accounts for the potential to exceed 6.3kW/ft.
CREV System intake flow rate	8,000 cfm	8,000 cfm +/- 10%	Sensitivity analyses were performed to determine worst-case scenario. The -10% value (7,200 cfm) is bounding and is used in the analysis.
CREV System recirculation flow rate	28,000 cfm	28,000 cfm +/- 10%	Sensitivity analyses were performed to determine worst-case scenario. The -10% value (25,200 cfm) is bounding and is used in the analysis.
CR Atmospheric Dispersion Factors, X/Qs (sec/m^3)	0-2 hr 3.40E-04 2-8 hr 2.25E-04 8-24 hrs 1.02E-04 24-96 hrs 7.16E-05 96-720 hrs 4.99E-05	0-2 hr 5.34E-04 2-8 hr 3.10E-04 8-24 hrs 1.36E-04 24-96 hrs 9.70E-05 96-720 hrs 6.02E-05	Analysis re-performed for equipment hatch, personnel hatch, emergency hatch, and station vent per RG 1.194. The highest of the 4 values was used in the analysis.

As shown in the enclosed Calculation No. C-1101-900-E000-083, Rev. 2, and tabulated below, the reanalyzed Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room operator dose consequences remain well below the acceptance criteria of 10 CFR 50.67 and GDC-19.

	Fuel Handling Accident Inside Containment TEDE Dose (Rem)		
	Control Room	**EAB**	**LPZ**
Current Licensing Basis Dose	6.55E-01	4.20E+00	7.35E-01
Re-Calculated Dose	1.12E+00	4.49E+00	7.87E-01
Allowable Dose	5.00E+00	6.30E+00	6.30E+00

2. **NRC Question**

Based upon a preliminary review of the proposed amendment the reviewer is unable to match the calculated doses for the accident analyses. It would be helpful if the licensee would provide their design-basis accident calculations. If the calculations are provided, answers to questions provided in this request for additional information (RAI) may reference the calculation.

Response

Refer to the response to Question No. 1 above.

3. **NRC Question**

The staff requests further information regarding the assumptions used to model the control room response to the fuel handling accident. Please provide the time dependent flow rates (filtered, unfiltered, recirculation, and pressurization), filtration efficiencies and time to isolate. Please provide justification for the values used.

Section 5.1.3 of Regulatory Guide 1.183 states that: "The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose . . . If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used." Confirm that the values used (with the tolerance bands (typically +/- 10%)) provide conservative control room doses.

Response

The nominal values for the Control Room Emergency Ventilation (CREV) System recirculation and intake flows are 28,000 cfm and 8,000 cfm, respectively. A table indicating the results of the sensitivity analysis, which considered all combinations of flow value tolerance levels, is provided on page 5 of Calculation No. C-1101-900-E000-083, Rev. 2 (Enclosure 2). As shown in this analysis, the worst-case tolerances were determined to be –10% on both the recirculation and intake flow values. The time dependent flow rates (filtered, unfiltered, recirculation, and pressurization), filtration efficiencies and time to isolate are included in the calculation and summarized below:

CREV System Recirculation:

25,200 cfm (28,000 cfm - 10%) from t = 30 minutes to 30 days
Filtration Efficiencies: 99% / 75% / 75% (particulate / elemental / organic)

Notes:
The 25,200 cfm represents only that portion of the flow through the fan/filtration unit that is recirculated. The 7,200 cfm (8,000 cfm -10%) intake flow is treated separately. These flow rates are conservative based on system testing. The particulate filter efficiency (99%) is per Technical Specification (TS) 3.15.1.2.a. The elemental and organic iodine adsorber efficiency (75%) used in the analysis conservatively bounds the ≥ 95% efficiency required by TS 3.15.1.2.b. It is assumed that the filtration unit does not operate before 30 minutes and continues to operate for the duration of the accident.

CR Filtered Intake:

4,000 cfm from t = 0 to 30 minutes (unfiltered)
7,200 cfm (8,000 cfm - 10%) from t = 30 minutes to 30 days
Filtration Efficiencies: 99% / 75% / 75% (particulate / elemental / organic)

Notes:
The 7,200 cfm represents that portion of the flow through the filtration unit that is taken into the control room via the intake tunnel (not recirculated yet). For conservatism, the 4,000 cfm represents one half of the normal intake flow for the 30-minute period during which the normal HVAC shuts off due to the isolation signal but before emergency ventilation is started. This value is added to the assumed unfiltered inleakage value of 1,000 cfm.

CR Unfiltered Inleakage:

1,000 cfm from t = 0 to 30 days (unfiltered)

This is a conservative assumed value with ample margin above the values measured using a tracer gas (i.e., worst case control room ventilation train "A" measured at 233 ± 129 scfm in the emergency mode), consistent with the existing licensing basis. This inleakage is assumed for the entire duration of the accident.

Time to Isolate CR:

30 minutes

This represents a conservative time to isolate and place the control room ventilation system in the emergency mode of operation, consistent with the existing licensing basis. These operator actions are performed within the Main Control Room on the H&V Panel.

4. **NRC Question**

Please provide the results of the dose analysis used to support the license amendment.

Response

The revised dose analysis used to support the proposed amendment is described in the response to Question No. 1 above, and is attached as Enclosure 2.

5. **NRC Question**

Regulatory Guide 1.183, Appendix B, Regulatory Position 1.2, states: "The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released."

a. Final Safety Analysis Report (FSAR) Section 14.2.2.1, "Fuel Handling Accident Analysis," states: "Because there is a chance that more than one spent fuel assembly may be damaged during refueling, the probability and consequences of dropping a spent fuel assembly in the core and damaging more fuel pins than the equivalent of one assembly was discussed (Reference 17). The conclusion is that the doses for failure of two assemblies would not be greater than the exposure guidelines of 10CFR 50.67 and no additional restrictions on fuel handling operations and plant operation procedures are needed." Based upon Table 14.2-5, "Postulated Fuel Handling Accident Dose Results (in the Reactor Building)," the Exclusion Area Boundary (EAB) Total Effective Dose Equivalent (TEDE) is 4.20 Rem TEDE. This dose is based upon 1 fuel assembly being damaged. If two fuel assemblies would be damaged, the dose would be double this value or 8.4 Rem TEDE based upon the FSAR analysis. The acceptance criterion given in Regulatory Guide 1.183 in Regulatory Position 4.4 for the EAB is 6.3 Rem TEDE. The amendment request proposes a relaxation of the containment integrity during fuel handling based upon a supporting analysis that potentially exceeds the Regulatory Guide 1.183 acceptance criteria. If the licensing basis of the fuel handling accident includes the damage of two assemblies, the staff does not understand why the resulting doses are not compared to the Regulatory Guide 1.183 acceptance criteria. Please provide justification why the two assemblies are not considered for the proposed design basis analysis supporting the proposed change.

b. FSAR Table 3.2-18, and Section 3.2.3.2.2.1 states that the design radial peaking factor for Three Mile Island, Unit 1 is 1.8. The January 23, 2001 license amendment request provides the parameters used in the analysis used as the justification for the proposed elimination of the equipment hatch. In the analysis that supports the January 23, 2001 amendment the radial peaking factor is given as 1.7. The NRC staff also used 1.7 in the safety evaluation dated October 2, 2001. TMI cites this safety evaluation as the basis for the proposed elimination of the containment equipment hatch during refueling. If the licensing basis for the design peaking factor is 1.8, justify why the previously approved analysis dated October 2, 2001 (that used a radial peaking factor of 1.7) is still valid for the proposed change. Please also explain why the FSAR value of 1.7 continues to be stated as the design basis value for the radial peaking factor for the fuel handling accident.

Response

a. The statement in UFSAR Section 14.2.2.1 regarding failure of two assemblies is an historical statement, and is based on an NRC Safety Evaluation Report (SER), dated December 31, 1979 which references a generic EG&G Idaho Technical Report study, dated October 1978 regarding the potential to damage more than one assembly in a postulated FHA. However, the TMI Unit 1 licensing basis FHAIC dose consequences have always been analyzed using the assumption of 208 damaged rods, one full assembly. This assumption of 208 damaged rods, one full assembly, has been maintained since the original plant licensing basis Safety Evaluation Report, dated July 11, 1973. The FHAIC alternative source term analysis supporting Amendment No. 236, dated October 2, 2001, and the revised FHAIC alternative source term analysis provided in Enclosure 2, assume 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. The fission product release from the breached fuel is based on Regulatory Guide 1.183, Appendix B, Position 1.2, and all gap activity in the damaged rods is assumed to be instantaneously released. Significant additional conservatism is added in the Enclosure 2 analysis by including two times the gap fractions specified in Regulatory Guide 1.183. Doubling the gap fraction has essentially the same effect as postulating two damaged assemblies. As shown in Enclosure 2, the radiological dose consequences remain well below the acceptance criteria of 10 CFR 50.67 and GDC-19.

   Additionally, it is noted that Reference 17 in UFSAR Section 14.2.2.1 is not a correct reference for this issue. This UFSAR error has been entered into the Corrective Action Program and TMI Unit 1 UFSAR Section 14.2.2.1 will be revised to accurately identify the appropriate analysis assumptions and parameters described in Enclosure 2.

b. The consequences of the Fuel Handling Accidents (FHA) analyzed in the TMI Unit 1 UFSAR are based on the source term (i.e., isotopic inventory) of an average power fuel assembly increased by a radial-local peaking factor of 1.70 to bound the highest powered assembly in the core. Radiological consequences of other design basis accidents are not impacted by design radial-local peaking factor. Subsequent to TMI Unit 1 Amendment No. 236, dated October 2, 2001, which implemented alternative source term for the FHA dose analysis, TMI Unit 1 Amendment No. 247, dated October 20, 2003, was issued to implement Statistical Core Design (SCD) methodology. The SCD methodology increased the design radial-local peaking factor for core thermal-hydraulic analyses from 1.714 to 1.80, which results in less restrictive axial imbalance core operating limits. The isotopic inventory of the fuel is a function of its steady-state power level. Higher peaking factors that may occur during transients are short-lived and have an insignificant impact on fuel isotopic inventories that accumulate over a two-year cycle. The maximum steady-state radial-local peaking factor (including physics model radial-local power uncertainty) for

TMI Unit 1 cycle designs applying the SCD methodology is 1.64, which is bounded by the radial-local peaking factor of 1.70 that is applied in the current TMI Unit 1 FHA analyses. Therefore the dose consequences of the current FHAs for TMI Unit 1 remain bounding for cycles designed with the SCD methodology.

The NRC explicitly agreed with this conclusion in Section 3.2 of the SER for Amendment No. 247, dated October 20, 2003. The NRC SER recognizes in Section 3.2 of this SER that the steady-state radial-local peaking factor is 1.64 for TMI Unit 1 Cycle15. Since this value was below the 1.70 limit used in the current FHA analyses, Amendment No. 247 concluded that the FHA for Cycle 15 is bounded by the previous analysis. Additionally, the TMI Unit 1 Cycle 16 (Fall 2005) core design preserves the maximum steady-state radial-local peaking factor of 1.64. The TMI Unit 1 steady-state radial-local peaking factor for each cycle design is implemented via the Cycle Design Inputs and Requirements (CDIR) document, which is controlled under 10 CFR 50.59 per the AmerGen core reload process. Any future change to the TMI Unit 1 cycle design implementing a steady-state radial-local peaking factor greater than 1.70 would result in an increase in the current FHA analyzed dose consequences described in the current UFSAR, and require prior NRC review and approval. UFSAR Section 14.2.2.1 will be revised to clarify that FHA analysis dose consequences are based on a limiting steady-state radial-local peaking factor of 1.70, as approved in Amendment No. 247, dated October 20, 2003.

6. <u>NRC Question</u>

What criteria will be used to determine if closure of the containment is necessary in the event of adverse weather? Has the impact of wind on fuel handling been evaluated (for example, reduced pool visibility due to pool surface disruption)? What steps would be taken in the event of severe weather to minimize the impact of flying debris or missile hazards?

<u>Response</u>

Existing procedures OP-AA-108-111-1001 and EP-1202-33 provide criteria applicable to the containment openings, including the equipment hatch, in the event of adverse weather, and to minimize the impact of debris and missile hazards. TMI site specific procedure EP-1202-33 actions are based on the following symptoms: (1) issuance of National Weather Service tornado/high wind storm watch or warning, and (2) site wind speed recordings of ≥ 50 mph. Additionally, Procedure EP-1202-33 requires operations and security personnel to monitor weather conditions and report changing conditions to the control room immediately. These symptoms provide time to take actions before conditions deteriorate. These procedures contain severe weather guidelines which direct the site to terminate fuel handling activities and to correct building integrity concerns due to open doors. Guidelines for high winds and tornados include consideration of actions to terminate all fuel movements in the storage pool and reactor cavity, to place any fuel being transferred into pool storage racks, to verify that all fuel assemblies on jib crane hooks and refueling grapples are removed and stored, and to verify all access doors from outside are secured closed including verification that the Reactor Building concrete missile shield is closed and evaluation of reinstalling the Reactor Building equipment hatch. Hurricane preparation guidelines include consideration of installing netting across the refueling cavity and spent fuel pools.

These procedure guidelines for severe weather essentially eliminate the potential impact of wind on fuel handling and the potential for reduced pool visibility. These guidelines also minimize the impact of flying debris or missile hazards.

Additionally it is noted that the containment equipment hatch opening (elev. 308'-0") and the Reactor Building refueling floor (elev. 346'-0") are separated by a concrete floor elevation. Since the equipment hatch opening is not in direct proximity to the refueling floor, it is unlikely that outside wind conditions will impact the pool surface visibility.

7. **NRC Question**

The August 22, 2001 RAI response stated that:

"AmerGen has performed a bounding assessment of the possible affects on the overall radiological dose results previously submitted. This assessment doubled the iodine release fraction previously used for additional conservatism to compensate for a higher peak pin power burnup and included the conservative assumptions from the original analysis. Particulate cesium and rubidium are retained by the water in the reactor cavity (per Regulatory Guide 1.183, B.3) and are not considered in this assessment. This is considered a bounding conservative assessment of the potential affects of the identified peak pin power burnup condition. The estimated results demonstrate only minimal potential impact, as defined by 10 CFR 50.59, on the previously calculated doses which remain well within the allowable dose criteria as specified in Regulatory Guide 1.183 and 10 CFR 50.67; and therefore, do not affect the original licensing basis analysis submitted on January 23, 2001."

Later in a February 15, 2002 RAI response stated "Additionally although the AmerGen letter 5928-01-20209 (August 22, 2001 letter) did not explicitly state that the noble gas release fractions were doubled, the bounding assessment referred to in that letter doubled halogen and noble gas release fractions as shown in the table above." The table above essentially doubled all the gap fractions used in the January 23, 2001 analysis.

The January 23, 2001, analysis provided the following table for the dose consequences (in TEDE) of the fuel handling accident in containment. These results appear to be the "overall radiological dose results previously submitted" that are referenced in the above August 22, 2001 quotation.

Dose	Control Room	EAB	LPZ
Calculated	6.55E-1	4.20E+0	7.35E-1
Allowable	5.00E+0	6.30E+0	6.30E+0

Based upon this table and the statements made above the staff requests additional clarification. If the August 22, 2001 bounding assessment doubled the iodine and noble gas release fractions then the January 23, 2001 dose analyses would have the following values:

Dose	Control Room	EAB	LPZ
Calculated	1.31E+0	8.40E+0	1.47E+0
Allowable	5.00E+0	6.30E+0	6.30E+0

The EAB doses appear to exceed the Regulatory Guide 1.183 acceptance criteria of 6.3 rem TEDE.

Response

The revised TMI Unit 1 FHAIC analysis of the control room and offsite doses, provided in Enclosure 2, includes two times the gap fractions for noble gases and iodines (minus cesium and rubidium which are retained in the water) per Regulatory Guide 1.183, Appendix B, Position 3, in conjunction with the revised bounding control room X/Q for the equipment hatch, personnel and emergency hatches, and the station vent (Reference AmerGen letter to the NRC, 5928-05-20124, dated June 30, 2005). As shown in Enclosure 2, the radiological dose consequences remain well below the acceptance criteria of 10 CFR 50.67 and GDC-19.

8. **NRC Question**

Based upon the above information, explain how the doses in the revised analysis stayed below the 6.3 Rem TEDE (given in Regulatory Guide 1.183) when the 1.183 gap fractions were doubled.

Response

The revised analysis referenced in the Question is superseded by the analysis provided in Enclosure 2. As described in the response to Question No. 7 above, the TMI Unit 1 FHAIC analysis of the control room and offsite doses includes two times the gap fractions for noble gases and iodines (minus cesium and rubidium which are retained in the water) per Regulatory Guide 1.183, Appendix B, Position 3. The dose analysis used to support the proposed amendment is further described in the response to Question No. 1 above, and is attached as Enclosure 2.

9. **NRC Question**

Based upon the descriptions provided above, the gap fractions in the fuel handling analysis appear to be double the values given in Regulatory Guide 1.183. Explain why the gap fractions provide in FSAR (Update-16, 4/02) Section, 14.2.2.1, "Fuel Handling Accident," state that the gap activity is based on Regulatory Guide 1.25 assumptions, i.e., 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85 and 10 percent of the total radioactive iodine in the rods at the time of the accident."

Response

This UFSAR Section will be revised to accurately describe the assumptions and parameters utilized in the analysis provided in Enclosure 2.

10. **NRC Question**

Regulatory Guide 1.183, Appendix A, Regulatory Position 5.3 states:

"The staff will generally require that technical specifications allowing such operations [allowing the airlocks or equipment hatch open] include administrative controls to close the airlock, hatch, or open penetration within 30 minutes."

TMI requested to use a temporary equipment hatch instead of replacing the equipment hatch. TMI states: "The contingency temporary hatch cover provides an atmospheric ventilation barrier to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored. The temporary equipment hatch cover is not intended to completely block the opening or be capable of resisting pressure. Therefore, the proposed change is consistent with the administrative controls applied to other previously approved containment openings." The bases state: "When a temporary equipment hatch cover is used in place of the equipment hatch, there are no special requirements for sealing, pressure retention, or complete blocking of the opening for this cover." Also, it states: "There are no special requirements to achieve continuous air flow into the Reactor Building." These words leave the flexibility to have no hatch in place and do not appear to provide a reliable defense-in-depth measure to block the flow of radiation in the event that a fuel handling accident occurs. Please provide additional justification why the proposed change meets the intents of the Regulatory Position to provide defense in depth against uncertainties in the radiological calculation given: 1) the purge system used to ensure the flow of the radiation is into the reactor building equipment hatch opening is not safety related, 2) the size of the replacement hatch is not defined, 3) the technical specification bases provide no requirements to achieve continuous air flow into the reactor building, and 4) the purge now may allow a flow path that emits more radioactivity into the environment than if no purge were used.

Response

The proposed Technical Specification change adds the containment equipment hatch opening to the existing TMI Unit 1 Technical Specification 3.8 administrative control requirements applicable to containment openings during the handling of irradiated fuel in the Reactor Building. These administrative controls ensure that appropriate personnel are aware that the equipment hatch is open, that a specific individual(s) is designated and available to close the equipment hatch opening as part of a required evacuation of containment, and that any obstruction that could prevent closure of the equipment hatch opening will be capable of being quickly removed.

The purge system is only being relied upon for "defense-in-depth." The calculation is performed without purge filtration credit. If exhaust flow is through the purge filters, the potential dose consequences analyzed will be further reduced.

The analysis was performed assuming that the equipment hatch is removed for the duration of the accident. The prompt closure of the opening provides a "defense-in-depth" measure, and is not credited in mitigating the accident. Continuous air flow into the building is another defense-in-depth feature that is not credited in the analysis. These measures are consistent with TSTF-51 guidance.

The revised dose analysis was performed using the assumption that all exhaust air (at the rate of 165,780 cfm to achieve a 2-hour release) from the containment will exit through the station vent without filtration since this location results in the bounding control room X/Q value.

The TMI Unit 1 proposed amendment request is based on previously approved temporary closure methods such as Crystal River 3 (NRC SER dated July 14, 2003), and the proposed TS Bases wording is based on TSTF-51 defense-in-depth guidance. Implementation of similar defense-in-depth guidance has been approved for other plants such as Fort Calhoun Station (NRC SER dated March 26, 2002), Sequoyah Nuclear Plant (NRC SER dated October 28, 2003), and Salem Nuclear Generating Station (NRC SER dated September 16, 2004). The TMI Unit 1 proposal is fully consistent with previously approved defense-in-depth provisions, as described by TSTF-51 guidance. The defense-in-depth provisions as implemented in TSTF-51 are not intended to mitigate *"uncertainties in the radiological analysis" as stated in the question. The purpose of the* defense-in-depth provisions is to further minimize dose consequences in the event of a FHAIC. The TMI Unit 1 temporary cover design will allow gaps between the cover and the walls of the opening. This design will satisfy the intent of the TSTF-51 guidance, which states that when a temporary cover is used in place of the equipment hatch, the cover "need not completely block the penetration or be capable of resisting pressure." The TSTF states further that the purpose of this "prompt method" of closing the hatch opening is to enable ventilation systems (which TMI has committed to maintain in operation during fuel movement) to draw the release from a postulated FHAIC in the proper direction such that it can be treated and monitored. All of this is to minimize potential dose consequences in the event of an FHAIC. These measures are not credited in any way in the dose analysis.

The TMI Unit 1 temporary cover is being designed to provide a contingency method of prompt closure of the containment equipment hatch opening when the permanent equipment hatch cover is fully removed and not capable of being closed within 45 minutes of containment evacuation due to the physical size and weight of the permanent hatch cover. Closure within 45 minutes of containment evacuation will provide reasonable assurance of prompt closure following an FHA even though containment closure is not required to meet acceptable dose consequences. An acceptance test will be performed to ensure that the temporary equipment hatch cover can be installed within 45 minutes to serve the intended function with the full design Reactor Building purge flow in effect. The 45-minute closure time is considered to begin when the control room communicates the need to shut the containment structure equipment hatch opening. This clarification is incorporated into the existing commitment.

Please note that as an alternative to the temporary cover design, use of the existing missile shield is being considered. If this alternative is selected, additional details regarding this approach will be provided by August 12, 2005.