October 13, 2005

Mr. Christopher M. Crane
President and Chief Executive Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: ELIMINATION OF CONTAINMENT EQUIPMENT HATCH CLOSURE REQUIREMENT DURING REFUELING (TAC NO. MC4904)

Dear Mr. Crane:

The Commission has issued the enclosed Amendment No. 257 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1, in response to your application dated October 20, 2004, as supplemented by letters dated June 30, July 29, August 17, and September 19, 2005.

The amendment revised the Technical Specifications to (1) eliminate the existing requirement in Section 3.8.6 regarding maintaining the containment equipment hatch cover in place with a minimum of four bolts during fuel loading and refueling operations, and (2) revise or introduce commitments to the Technical Specifications Bases in support of the change in Section 3.8.6.

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/

Peter S. Tam, 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. 257 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

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AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 257
License No. DPR-50

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

   A. The application for amendment by AmerGen Energy Company, LLC, et al., (the licensee), dated October 20, 2004, as supplemented by letters dated June 30, July 29, August 17, and September 19, 2005, 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. 257, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by J. Boska /

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 13, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 257

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

Remove	Insert
3-44	3-44
3-45	3-45
3-45a	3-45a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 257

TO FACILITY OPERATING LICENSE NO. DPR-50

AMERGEN ENERGY COMPANY, LCC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1)

DOCKET NO. 50-289

1.0 INTRODUCTION

By application dated October 20, 2004 (Agencywide Document Access and Management System (ADAMS) Accession No. ML042950487), as supplemented by letters dated June 30, July 29, August 17, and September 19, 2005 (Accession Nos. ML051940495, ML052220637, ML052350562, and ML052660214, respectively), AmerGen Energy Company, LLC (the licensee) requested changes to the Technical Specifications (TSs) for TMI-1. The licensee proposed to revise the TSs to (1) eliminate the existing requirement in Section 3.8.6 regarding maintaining the containment equipment hatch cover in place with a minimum of four bolts during fuel loading and refueling operations, and (2) revise or introduce commitments to the TSs Bases in support of the change in Section 3.8.6. The supplements cited above provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 7, 2004 (69 FR 70714).

2.0 REGULATORY EVALUATION

The licensee described the proposed amendment and provided its technical analyses and regulatory analyses in Sections 2.0, 4.0, and 5.0 of Enclosure 1 to the licensee's October 20, 2004, application. The supplements described above clarified the application.

The NRC staff finds that the licensee identified applicable regulatory requirements in Section 5.2 of its submittal. The proposed amendment would allow the equipment hatch to be open during refueling operations when there is handling of irradiated fuel assemblies inside the Reactor Building. Based on this, the proposed amendment necessitates the NRC staff's evaluation of the licensee's design basis fuel-handling accident inside containment (FHAIC), and containment integrity (i.e., the equipment hatch is part of the containment pressure boundary) during handling of irradiated fuel.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 19, "Control Room," requires that licensees maintain the control room in a safe condition under accident conditions. Under these conditions, the licensee must provide adequate radiation protection to permit access and occupancy of the control room. Requirements in 10 CFR 50.67, “Accident Source Term,” on the other hand, establish the dose limits for the exclusion area and for the low population zone.

To show that the radiation doses, onsite and offsite, will meet the above regulatory requirements, licensees have performed evaluations of their accident radiation doses. The NRC staff provides guidance for these evaluations in regulatory guides (RGs). The regulatory requirements on which the NRC staff based its review are contained in 10 CFR Part 50, GDC 19 of Appendix A, and 10 CFR 50.67. Except where the licensee proposed a suitable alternative, the NRC staff used the regulatory guidance provided in RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” in performing this review.

In addition to reviewing the application and supplements, the NRC staff also considered relevant information in the TMI-1 Final Safety Analysis Report - Updated Version (UFSAR), TMI-1 TSs, and the licensee's responses to Generic Letter (GL) 2003-01, “Control Room Habitability.”

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's proposed changes, and regulatory and technical analyses in support of the proposed amendment. These are contained in Sections 2.0, 4.0, and 5.0 of Enclosure 1 to the licensee's October 20, 2004, application, and supplements.

As stated in its application, the licensee proposes to add the following note to TS 3.8.6:

> The equipment hatch may be open if all of the following conditions are met:
>
> 1. The Reactor Building Equipment Hatch Missile Shield Barrier is capable of being closed within 45 minutes,
> 2. A designated crew is available to close the Reactor Building Equipment Hatch Missile Shield Barrier, and
> 3. Reactor Building Purge Exhaust System is in service.

Concurrently, the licensee proposed to amend the Bases of the TSs to include: (1) an explanation of the new note added to the TSs; (2) controls to bypass the Reactor Building purge valve high radiation interlock; (3) a provision to ensure the capability to communicate with the control room during fuel handling; and (4) a test of the reactor building purge valves. The test will determine that the Reactor Building purge valves will remain open when the isolation system is bypassed.

The licensee stated that the proposed change is to permit the containment equipment hatch to remain open during “REFUELING SHUTDOWN” or “REFUELING OPERATIONS.” TS 3.8.6 does not currently permit this. As discussed in Section 3.2 below, in support of the requested change, the licensee proposed to add related administrative controls.

The postulated accident that could result in a release of radioactive material through the equipment hatch would be an FHAIC, as discussed in the evaluation below.

3.1 Postulated Accidents

The licensee stated that the limiting design-basis event during refueling when there are core alterations or movement of irradiated fuel inside containment is the FHAIC. The licensee has described this event in Section 14.2.2.1 of the UFSAR, and the licensee's October 20, 2004, application and supplements. RG 1.183 provides the acceptance criteria pertaining to the evaluation of this accident.

3.1.1 FHAIC Radiological Consequence Analysis

In TMI-1's FHAIC analysis, the fuel rods are assumed to rupture, releasing the radionuclides within the fuel rod to the reactor cavity water. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released as a result of the accident. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity depending on their physical and chemical form. The licensee assumed no decontamination for noble gases, an overall effective decontamination factor of 200 for radioiodines, and retention of all particulate fission products. The licensee also assumed that essentially 100 percent of the fission products released from the reactor cavity are released to the environment in 2 hours without any credit for filtration.

The licensee calculated potential dose consequences for the FHAIC at the exclusion area boundary (EAB), the low-population zone (LPZ) and the control room. The assumptions used for the calculated dose consequences are summarized in Tables 1, 2, and 3, of this safety evaluation (SE).

The TMI-1 radiological analysis supporting alternative source term implementation for the FHAIC was previously approved in TMI-1 Amendment No. 236, dated October 2, 2001 (Accession No. ML0127102140. This analysis is contained in AmerGen Calculation No. C-1101-900-E000-083, Rev. 1, dated January 19, 2001. The licensee did not revise this analysis to support the proposed change for the containment equipment hatch originally submitted in the October 20, 2004, application. However, the licensee subsequently revised Rev. 1 of this calculation to address NRC questions contained in a request for additional information (RAI). The changes alter: (1) the fraction of fission products assumed, (2) the control room emergency ventilation system flow rates, (3) the atmospheric dispersion factors, and (4) the control room and offsite doses. The NRC staff's evaluation of Changes 1 and 2 are contained in the current Section 3.1.1. Change 3 is addressed in Section 3.3.1.1, while Change 4 is evaluated in Sections 3.1.2 and 3.1.3 below.

The licensee's proposed revised analysis uses "two times the iodine and noble gas values without contribution due to cesium." In a letter dated February 15, 2002 (ADAMS Accession No. ML020640340), the licensee provided gap release fractions for fuel assemblies exceeding 54 GWD/MTU (gigawatt days per metric ton uranium) burnup and 6.3 kw/ft linear heat generation rate. This letter stated that the licensee doubled both the iodine and noble gas fractions provided in RG 1.183 for the FHAIC. The NRC staff reviewed these doubled gap fractions previously and found them acceptable (see letter from T. Colburn, dated April 30, 2002, ADAMS Accession No. ML021080289). With respect to the modeling of the cesium, the NRC staff compared this modeling with RG 1.183. Appendix B, Regulatory Position 1.2 of RG 1.183, states that cesium should be considered when modeling the fuel-handling accident. Regulatory Position 3 states that particulate radionuclides are assumed to be retained by the reactor cavity. The NRC staff, therefore, finds that assuming that the cesium is not released from the reactor cavity is consistent with the regulatory positions of RG 1.183. Therefore, the doubling of the iodine and noble gas fractions (as compared to RG 1.183) and the proposed modeling of cesium are acceptable.

The licensee considered the dose to control room operators as a result of the FHAIC. In its October 20, 2004, application, the licensee assumed that the control room unfiltered in-leakage is 1,000 cubic feet per minute (cfm), both before and after the control room is manually isolated. The NRC staff requested additional information about the licensee's use of 1000 cfm for unfiltered inleakage before the manual actuation. In its September 19, 2005, response, the licensee changed this assumption to a value of 61,000 cfm of unfiltered inleakage for the design basis FHAIC. The licensee stated that a sensitivity study has shown that at values greater than 61,000 cfm the control room operator dose is insensitive to additional unfiltered inleakage and, therefore, a test to determine the actual unfiltered inleakage in the normal mode is not required. The NRC staff performed its own sensitivity evaluation and confirmed the results of the TMI-1 sensitivity study. Therefore, the NRC staff finds the use of 61,000 cfm for the design basis analysis, as described in this SE, is acceptable.

3.1.1.1 Atmospheric Dispersion Estimates

The licensee calculated new atmospheric dispersion factors (χ/Q values) for use in evaluating the impact of an FHAIC on the TMI-1 control room. The resulting TMI-1 control room χ/Q values represent a change from those currently presented in Chapter 2.5.4.2.1 of the TMI-1 UFSAR. The licensee used previously approved EAB and LPZ χ/Q values listed in Chapter 2.5.4.1 of the TMI-1 UFSAR to perform offsite dose assessments related to an FHAIC.

3.1.1.1.1 Meteorological Data

The licensee generated the new control room χ/Q values for this amendment application using site meteorological data collected during 1992, 1993, 1995, and 1996. The combined data recovery of wind speed, wind direction, and stability (delta-temperature) exceeded 90 percent for each of these 4 years. The licensee provided these hourly data in its letter dated June 30, 2005. The licensee excluded the 1994 meteorological data from the atmospheric dispersion analysis because the combined data recovery for this year was below 90 percent.

The NRC staff previously accepted the 1992, 1993, 1995, and 1996 meteorological data for use in atmospheric dispersion analysis with its approval of the χ/Q values discussed in the SEs associated with TMI-1 License Amendments Nos. 210 and 215, dated April 15, 1999, and August 24, 1999, respectively. Nonetheless, the NRC staff performed a quality review of the 1992, 1993, 1995, and 1996 hourly meteorological database provided by the licensee using the methodology described in NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data.” Further review was performed using computer spreadsheets. Wind speed and wind direction were measured on the onsite meteorological tower at a height of 100 feet (30.5 meters) above the ground. Stability class was calculated using the temperature difference between the 150-foot (45.7-meter) and 33-foot (10-meter) levels. As expected, the NRC staff's examination of the data revealed generally stable and neutral atmospheric conditions at night and unstable and neutral conditions during the day. Wind speed, wind direction, and stability class frequency distributions were reasonably similar from year to year.

For the reasons cited above, the NRC staff concludes that the 1992, 1993, 1995, and 1996 meteorological database provides an acceptable basis for making atmospheric dispersion estimates for use in the FHAIC dose assessments performed in support of this application for amendment.

3.1.1.1.2 Control Room Atmospheric Dispersion Factors

In its October 20, 2004, application the licensee relied on the previously approved FHAIC control room dose consequence analysis associated with Amendment No. 236 dated October 2, 2001. The dose calculated for the FHAIC radiological consequence analysis for Amendment No. 236 used χ/Q values that were derived assuming a diffuse source release from the reactor building; i.e., the activity being released was assumed to be homogeneously distributed throughout the reactor building and leak over the plane of the reactor building surface perpendicular to the control room air intake tunnel. In reality, the releases are likely to occur through the unit vent or through an open personnel hatch, equipment hatch, or emergency air lock. The NRC staff asked the licensee in Question 1 of the RAI letter dated May 2, 2005, to confirm that the diffuse source control room χ/Q values used in the Amendment No. 236 dose assessment bound the atmospheric dispersion characteristics for releases from the unit vent, personnel hatch, equipment hatch, and emergency air lock.

In its RAI response dated June 30, 2005, the licensee presented new control room χ/Q values for releases from the unit vent, personnel hatch, equipment hatch, and emergency air lock. The highest set of χ/Q values from these four release pathways (which was associated with unit vent releases) was then used in the revised control room dose consequence analysis for the FHAIC. The licensee used guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," to generate these new control room atmospheric dispersion factors. The new control room χ/Q values were calculated using the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in design-basis accident radiological analyses.

The licensee executed ARCON96 using the 1992, 1993, 1995, and 1996 hourly data from the site meteorological tower. Wind speed and wind direction data from the site tower's 100-foot (30.5-meter) level were provided as input and stability class was calculated using the temperature difference between the 150-foot (45.7-meter) and 33-foot (10-meter) levels. Each of the release pathways (i.e., the unit vent, personnel hatch, equipment hatch, and emergency air lock) were modeled as point sources using the ARCON96 ground-level release mode option. All distances from the release points to the control room air intake were minimized. In the case of personnel hatch releases, a "taut string length" was used to account for the intervening fuel handling building.

The licensee generated the control room χ/Q values using the control room yard air intake as the receptor. The FHAIC radiological consequence analysis assumes all infiltration into the control room (both filtered and unfiltered) occurs through the control room yard air intake. The NRC staff asked the licensee in Question 2 of the May 2, 2005, RAI letter to confirm that there are no potential unfiltered inleakage pathways during the control room isolation mode that could result in χ/Q values that are higher than the control room yard air intake χ/Q values. In its June 30, 2005, response, the licensee stated that tracer gas testing confirmed that there are no specifically identified potential unfiltered inleakage pathways in the control room isolation mode that could result in χ/Q values that are higher than that for the control room yard air intake. This was further demonstrated by the fact that differential pressure readings between all areas within the control room envelope and outside areas were shown to be positive. This implies that the most credible source of inleakage would be through the yard air intake tunnel.

The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the TMI-1 site. The NRC staff qualitatively reviewed the input data to the ARCON96 calculations and found them generally consistent with site configuration drawings and NRC staff practice. The NRC staff made an independent evaluation of the resulting atmospheric dispersion estimates by running the ARCON96 computer code and obtained similar results.

In summary, the NRC staff reviewed the licensee's assessments of control room post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The resulting control room χ/Q values are presented in Table 2. On the basis of this review, the NRC staff concludes that the χ/Q values presented in Table 2 are acceptable for use in the FHAIC control room dose assessments performed in support of the proposed amendment.

3.1.1.1.3 Offsite Atmospheric Dispersion Factors

The licensee evaluated offsite doses using the EAB and LPZ χ/Q values provided in the TMI-1 UFSAR Chapter 2.5.4.1. These values, presented in Table 3, were calculated using a methodology described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and site meteorological data from the years 1992, 1993, 1995, and 1996. The NRC staff has previously reviewed and found these χ/Q values acceptable as discussed in the SE associated with Amendment No. 210 dated April 15, 1999.

The NRC staff reviewed the licensee's use of the existing TMI-1 EAB and LPZ χ/Q values and found them to be appropriate for the application in which they are being used. On the basis of this review, the NRC staff concludes that the EAB and LPZ χ/Q values presented in Table 3 are acceptable for use in the FHAIC offsite dose assessments performed in support of the proposed amendment.

3.1.2 Control Room Doses and Unfiltered Inleakage

The NRC staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room inleakage rates assumed by licensees in analyses of control room habitability. The NRC staff issued GL 2003-01, "Control Room

Habitability," on June 12, 2003. The licensee responded to this GL by letter dated December 9, 2003. In its response, the licensee reported that inleakage testing using the American Society for Testing and Materials E741 tracer gas methodology determined a control room unfiltered inleakage rate of 233 +/-129 scfm for the "A" ventilation train and 189 +/-103 scfm for the "B" ventilation train. Table 1 provides the proposed values assumed for the FHAIC. These values are larger than the measured values reported in the licensee's tracer gas test results.

The NRC staff is still reviewing TMI's December 9, 2003, response for final resolution of GL 2003-01. However, the NRC staff has determined that there is reasonable assurance that the TMI-1 control room would be habitable during the design basis FHAIC, and that an evaluation of the licensee's current amendment application can be made before final resolution of the generic issue. The NRC staff made this determination based on (1) the results of the tracer gas testing at TMI-1, (2) the independent confirmatory calculations performed by the NRC staff, (3) the available margin between the licensee's FHAIC assumed inleakage and the actual measured inleakage (after 30 minutes), and (4) the use of the 61,000 cfm for the normal mode (0-30 minutes) as described in Section 3.1.1 above. This SE's finding that TMI-1's assumptions for control room doses and unfiltered inleakage during the FHAIC are acceptable is limited to only the scope of this amendment. As the NRC staff continues its review of GL 2003-01, additional information may be necessary to supplement the licensee's December 9, 2003, response letter. The NRC staff will address any future resolution in separate correspondence once review of the generic issue for TMI-1 is complete.

3.1.3 Offsite Doses

The EAB and LPZ doses estimated by the licensee for the FHAIC were found to be acceptable. The NRC staff performed independent calculations using the licensee's assumptions and confirmed the licensee's conclusions.

3.2. Controls on the Equipment Hatch Penetration

The licensee has proposed to allow the Reactor Building equipment hatch to be open during "REFUELING SHUTDOWN" or "REFUELING OPERATIONS." However, the missile shield doors would be maintained in an isolable condition (i.e., capable of being closed) and controls would be required as stated in the following proposed TSs:

- The Reactor Building equipment hatch missile shield barrier is capable of being closed within 45 minutes.

- A designated crew is available to close the Reactor Building Equipment Hatch Missile Shield Barrier.

- The Reactor Building purge exhaust system is in service.

- Administrative controls shall ensure that the Reactor Building purge system is in service, 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., cable and hoses) that could prevent closure of the air lock door or other penetration will be capable of being quickly removed.

The following existing TSs provide additional controls:

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

- TS 3.8.11 - 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, as determined by a shiftly check and daily verification.

The licensee also provided the following commitments:

- Procedural controls will ensure that during the movement of irradiated fuel, the equipment hatch/missile shield area will be manned 24 hours/days, 7 days/week in support of the outage unless the equipment hatch is closed and 4 bolts are installed.

- Complete permanent installation of steel plate to the lower most missile shield carriage area. The added plate will cover the area where grating is currently installed.

- Prior to initial use of this revised TS (i.e., TS 3.8.6), the licensee will demonstrate that the 45-minute closure duration is achievable.

- TMI Unit 1 procedures will include the following requirements to ensure that GDC 64 requirements will continue to be met during the movement of irradiated fuel:

    1) If the Reactor Building equipment hatch is removed (open), then place the purge system in operation and control the air flow at the hatch so that the prevailing continuous direction of air flow is into the Reactor Building.

    2) If the condition, as described in item 1 above, cannot be maintained, then fuel handling operations will be terminated until the Reactor Building equipment hatch is closed or purge is restored.

    3) Whenever the purge system is operating, ensure purge exhaust radiation monitor is operable or obtain periodic samples as currently specified in the Offsite Dose Calculation Manual (ODCM).

    4) Whenever the hatch is open, position a portable radiation monitor at the Reactor Building equipment hatch opening.

    5) If the purge system is operated with the Reactor Building equipment hatch open, bypass the Reactor Building purge exhaust high radiation interlock.

    6) Prior to initiating irradiated fuel movement with the Reactor Building equipment hatch open, verify the purge system is operating.

In addition to the controls and commitments the licensee provided above, the licensee provided the following clarifying and additional information.

- Each individual's (of the designated crew responsible for closure of the equipment hatch opening) identified duties will be in support of the loading and unloading of outage equipment at the hatch area.

- The total area of the clearances between the shield door and containment with the Reactor Building purge exhaust system operating and the missile shield installed ensure that the flow is into the Reactor Building.

- The process of closing the missile shield barrier is not impacted by adverse weather conditions. The ability to close the missile shield barrier is not impacted by operation of the Reactor Building purge exhaust system.

The licensee states that the above described TS revisions, along with existing TSs controls, provide assurance that the intent of closure as a defense-in-depth measure is accomplished. The licensee also states that the Reactor Building equipment hatch missile shield barrier provides an atmospheric ventilation barrier to enable ventilation systems to draw the release from a postulated FHAIC in the proper direction such that it can be treated and monitored.

Allowing the equipment hatch penetration to be open during refueling is partially compensated by the licensee implementing the proposed TS revisions, along with existing TS controls, and the commitments and clarifications cited above. RG 1.183, Appendix B, Regulatory Position 5.3 states that the NRC staff will generally require that TSs allowing such operations (as an equipment hatch open during refueling) include controls to close the open penetration. The proposed closure of the equipment hatch shield doors does not provide isolation as the equipment hatch does since areas of the penetration will remain open. With the shield doors closed, gaps between the shield door and containment will exist. The licensee proposed to establish an additional control to ensure that the airflow is in the proper direction. The control that the Reactor Building purge exhaust system is in service provides this assurance. RG 1.183 also states that "[s]uch administrative controls will generally require that a dedicated individual be present, with necessary equipment available to restore containment closure should a fuel handling accident occur." The NRC staff finds that the proposed controls meet the intent of this regulatory position.

Based upon the assessment that the proposed controls, commitments, and clarifications meet the intent of Regulatory Position 5.3, the NRC staff finds them acceptable. The NRC staff determined that these controls and commitments provide an important element of defense-in-depth, and with these controls and commitments in place, they will assure that the licensee will manage the consequences of an FHAIC in a manner that will afford adequate protection to the public.

3.2.1 Tornado Missiles

The missile shield doors and the equipment hatch provide missile protection for inside the containment. The missile shield covers the equipment hatch. The equipment hatch shield doors are designed for protection against generated missiles.

In addressing what will happen on the site during refueling with severe weather in the vicinity of the plant, the licensee stated in its July 29, 2005, letter that existing procedures are in place to provide criteria applicable to the containment openings, including the equipment hatch, in the event of adverse weather to minimize the impact of debris and missile hazards. These procedures contain severe weather guidelines. These guidelines direct licensee personnel to terminate fuel handling activities and to correct building integrity due to open doors in the event of severe weather. 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. Additionally, 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.

3.3 Other Considerations

The NRC staff identified several issues that complicated the review of this license amendment, including several inconsistencies in the licensing basis documents that describe the TMI-1 FHAIC. These issues are documented below for the licensee's consideration and corrective actions.

3.3.1 Inconsistencies Between July 29, 2005, Response and the Previous FHAIC

The subject response is not consistent with previously docketed information. The July 29, 2005, RAI response states that the previous analysis (Calculation No. C-1101-900-E0000-083, Revision 1) was revised to double the noble gas values assumed as the fraction of fission product inventory in the gap. The new calculation (Revision 2) states that the noble gas values assumed are 0.20 for Kr-85 and 0.10 for all other noble gases. Therefore, according to the July 29, 2005 response, the previous calculation (Rev. 1 - as stated in the July 29, 2005, letter) should have been 0.10 for Kr-85 and 0.05 for all other noble gases. Yet, the licensee's February 15, 2002, letter (based upon Rev. 1 of the calculation) provided the values for Kr-85 as 0.20 and other noble gases as 0.10. It appears that the information provided in the February 15, 2002, letter may not be correct.

3.3.2 UFSAR Contains Errors and Is Not Up-to-Date

The NRC staff noted inconsistencies between the license application documents and the UFSAR. Because of these inconsistencies, an RAI was issued on June 30, 2005 (Accession No. 051810253). The July 29, 2005, response to Question 5a of the June 30, 2005, RAI stated that a reference in the UFSAR was not correct and that the TMI-1 UFSAR Section 14.2.2.1 will be revised to accurately identify the appropriate analysis assumptions and parameters. The licensee’s July 29, 2005, response to Question 9 stated that the gap fractions contained in UFSAR Section 14.2.2.1 (Revisions 16 and 17, dated April 2002 and April 2004, respectively) had not been updated to incorporate the approved alternative source term analysis (approved October 2, 2001). For example, bulleted items d and f provide incorrect assumptions for the pool decontamination factor and the gap fractions.

3.4 Summary of Technical Evaluation

As delineated above, the NRC staff reviewed the assumptions, input parameters, and methods used by the licensee to assess the impacts of the proposed change to the TMI-1 TSs. Based on its review, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0, above. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the control room doses would continue to comply with GDC 19 (i.e., 5 rem total effective dose equivalent (TEDE)). The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB and LPZ doses would continue to be well within 10 CFR Part 50.67 (i.e., 6.3 rem TEDE). Therefore, the proposed license amendment is acceptable with regard to the radiological consequences of the postulated fuel-handling accident.

The NRC staff has reviewed the description of the proposed controls and commitments in the licensee's application and concludes that these descriptions are acceptable. The implementation of these controls and commitments will provide additional assurance of safety. Controls and commitments have also been added to the TS Bases.

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 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 (69 FR 70714). 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.

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Date: October 13, 2005

Table 1 (sheet 1 of 2)
Parameters and Assumptions Used in Analysis of FHAIC

Core thermal power, MWt	2619
Time between plant shutdown and accident, hrs.	72
Fraction of fission product in fuel element gap	
I-131	0.16
Kr-85	0.20
Other Noble Gases	0.10
Other Halogens	0.10
Fraction of gap activity released to the refueling cavity water, %	100
Radial peaking factor	1.7
Damaged fuel rods	208
Number of fuel assemblies in core	177
Minimum water depth above reactor vessel flange (and damaged fuel rods), ft	23
Refueling Cavity Water Decontamination Factors:	
Noble Gases	1
Iodine	200
Airborne Iodine Forms, %	
Elemental	57
Organic	43
Duration of release from containment, hrs.	2
Activity release rate from containment, ft^3/min	165,780
EAB breathing rate, m^3/sec	3.47E-4
EAB occupancy factor (0-2 hrs.)	1.0
LPZ breathing rate (m^3/sec)	
0-8 hr.	3.5E-4
8-24 hr.	1.8E-4
24-720 hr.	2.3E-4

Table 1 (sheet 2 of 2)
Parameters and Assumptions Used in Analysis of FHAIC

Control Room Parameters

Control room breathing rate m^3/sec	3.47E-4
Control room occupancy factor (0 - 24 hrs.)	1.0
Control room volume, ft^3	250,000
Unfiltered ingress/egress rate, ft^3/min.	10
Control room manual isolation, min.	30
Control room HVAC system operation (0 - 30 min.):	
Normal operation unfiltered inflow rate plus unfiltered inleakage rate, ft^3/min	61,000[1]
Control Room CREV System Operation (30 min. - 30 days):	
Filtered inflow rate, cfm	8000±10%
Filtered recirculation flow rate, ft^3/min.	28,000±10%
Unfiltered inleakage rate, ft^3/min.	1000[1]
Filter Efficiencies (iInflow and recirculation), %	
Elemental iodine	75
Organic iodide	75

[1] This value includes 10 ft^3/min that is assumed for unfiltered inleakage due to ingress and egress.

Table 2
Control Room Atmospheric Dispersion Factors Used in Analysis of FHAIC

Time Interval	χ/Q Value (sec/m^3)
0–2 hrs	5.34×10^{-4}
2–8 hrs	3.10×10^{-4}
8–24 hrs	1.36×10^{-4}
24–96 hrs	9.70×10^{-5}
96–720 hrs	6.02×10^{-5}

Table 3
Offsite Atmospheric Dispersion Factors Used in Analysis of FHAIC

Receptor	Time Interval	χ/Q Value (sec/m^3)
EAB	0–2 hrs	8.0×10^{-4}
LPZ	0–2 hrs	1.4×10^{-4}
	2–8 hrs	6.0×10^{-5}
	8–24 hrs	3.9×10^{-5}
	24–96 hrs	1.6×10^{-5}
	96–720 hrs	4.0×10^{-6}

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