March 17, 2003

Mr. Michael Kansler
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: SELECTIVE ADOPTION OF ALTERNATE SOURCE TERM (TAC NO. MB5382)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 215 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated June 5, 2002, as supplemented on January 9 and March 4, 2003.

The amendment revises the TSs to implement the alternate source term methodology for the fuel-handling accident analysis. Specifically, the amendment revises TS 3.9.3, "Containment Penetrations," to: (1) permit the equipment closure hatch opening and the personnel air lock doors to be capable of being closed during movement of irradiated fuel, (2) allow use of administrative controls for unisolating containment penetrations during movement of irradiated fuel, (3) delete the containment purge and containment pressure relief requirements and associated surveillances with the reactor subcritical for less than 550 hours, and (4) eliminate the TS applicability "during core alterations." In this regard, the amendment adopts TS Task Force (TSTF) Standard TS Change Travelers TSTF-68, "Containment Personnel Airlock Doors Open During Fuel Movement," TSTF-312, "Administratively Control Containment Penetrations," and, in part, TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations." The amendment also revises the Applicability Statements for Limiting Condition for Operation (LCO) 3.3.8 for the fuel storage building emergency ventilation system (FSBEVS) actuation instrumentation and LCO 3.7.13 for the FSBEVS to also add the term "recently" before "irradiated fuel assemblies." In addition, the LCO Required Action would likewise be modified to add the term "recently" to now require the suspension of movement of recently irradiated fuel in the FSB.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 215 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 215 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

TSs: ML
Package Number.: ML
Accession Number: ML030760135 *See previous concurrence

OFFICE	PDI-1\PM	PDI-1\LA	SPSB\SC*	SPLB\SC*	RORP\TSTF*
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DATE	03/13/03	03/13/03	03/06/03	03/06/03	03/05/03

OFFICE	OGC*	PDI-1\SC			
NAME	AHodgdon	RLaufer			
DATE	03/12/03	03/14/03			

Official Record Copy

DATED: March 17, 2003

AMENDMENT NO. 215 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT UNIT 3

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ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 215
License No. DPR-64

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

   A. The application for amendment by Entergy Nucleaar Operations, Inc. (the licensee) dated June 5, 2002, as supplemented on January 9 and March 4, 2003, 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-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 215, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 17, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 215

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

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

Remove Pages	Insert Pages
3.3.8-1	3.3.8-1
3.7.13-1	3.7.13-1
3.9.3-1	3.9.3-1
3.9.3-2	3.9.3-2
3.9.3-3	3.9.3-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated June 5, 2002, as supplemented on January 9 and March 4, 2003, Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs). The requested changes would revise the TSs to implement the alternate source term (AST) methodology for the fuel-handling accident (FHA) analysis. Specifically, the proposed amendment would revise TS 3.9.3, "Containment Penetrations," to: (1) permit the equipment closure hatch access door opening and the personnel airlock doors to be capable of being closed during movement of irradiated fuel, (2) allow use of administrative controls for unisolating containment penetrations during movement of irradiated fuel, (3) delete the containment purge and containment pressure relief requirements and associated surveillances with the reactor subcritical for less than 550 hours, and (4) eliminate the TS applicability "during core alterations." In this regard, the proposed amendment would adopt TS Task Force (TSTF) Standard TS Change Travelers TSTF-68, "Containment Personnel Airlock Doors Open During Fuel Movement," TSTF-312, "Administratively Control Containment Penetrations," and, in part, TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations." The amendment would also revise the Applicability Statements for Limiting Conditions for Operation (LCO) 3.3.8 and LCO 3.7.13 to also add the term "recently" before "irradiated fuel assemblies." In addition, the LCO Required Action would likewise be modified to add the term "recently" to now require the suspension of movement of recently irradiated fuel in the fuel storage building (FSB). The January 9 and March 4 letters provided clarifying information that did not expand the scope of the proposed amendment or change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff finds that the licensee in its June 5, 2002, application, as supplemented on January 9 and March 4, 2003, identified the applicable regulatory requirements. The regulatory requirements and guidance which the staff considered in its review of the requested action are as follows:

1. Section 50.67, "Accident source term," of Title 10 of the *Code of Federal Regulations* (10 CFR 50.67) provides the requirements to any licensee seeking to revise its current accident source term in design basis radiological consequence analyses.

2. General Design Criterion (GDC) 19, “Control room,” of Appendix A to 10 CFR Part 50 requires, in part, that a control room be provided with adequate radiation protection to permit access and occupancy under accident conditions.

3. GDC 56, “Primary containment isolation,” requires that each line that connects directly to the containment atmosphere and penetrates primary containment shall be provided with containment isolation valves.

4. NRC-approved Generic Changes to the Standard TS:

   a. TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,” November 1, 1999;

   b. TSTF-68, Revision 2, “Containment Personnel Airlock Doors Open During Fuel Movement,” August 16, 1999; and

   c. TSTF-312, Revision 1, “Administratively Control Containment Penetrations,” August 16, 1999

3.0 TECHNICAL EVALUATION

3.1 Background

The licensee submitted this license amendment as a selective implementation of the AST, as described in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” pursuant to 10 CFR 50.67, “Accident Source Term.” The current IP3 accident source term was developed using Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor Sites.” The proposed changes would revise the radiological consequence analyses for the postulated FHA allowing the equipment hatch and the personnel airlock doors to be open during refueling operations.

3.2 Proposed TS Changes

The specific changes requested by the licensee are as follows:

1. LCO 3.9.3.a in TS Section 3.9.3, “Containment Penetrations,” would be revised to allow the equipment closure hatch personnel access door to be capable of being closed during movement of irradiated fuel.

2. LCO 3.9.3.b in TS Section 3.9.3, “Containment Penetrations,” would be revised to allow the personnel air lock doors to be capable of being closed during movement of irradiated fuel.

3. LCO 3.9.3.c in TS Section 3.9.3, “Containment Penetrations,” would be revised to allow the use of administrative controls for not isolating containment penetrations during movement of irradiated fuel.

4. LCO 3.9.3.d and LCO 3.9.3.e in TS Section 3.9.3, “Containment Penetrations,” would be deleted along with its surveillance requirements regarding containment purge relief requirements with the reactor sub-critical for less than 550 hours.

5. APPLICABILITY in TS Section 3.9.3, “Containment Penetrations,” would be revised to delete “during core alteration.”

In its March 4, 2003, letter, the licensee withdrew its original request to relocate the requirements for the fuel storage building emergency ventilation system (FSBEVS) and FSBEVS Actuation System to the Technical Requirements Manual.

The licensee adopted the Improved Technical Specifications (ITS) in License Amendment No. 205 dated February 27, 2001, based on NUREG-1431, "Standard Technical Specifications [STS] for Westinghouse Plants," Revision 1, dated April 1995. Since then, industry and the NRC staff have been working to improve the ITS, in NUREG-1430 through NUREG-1434 for the different plant vendors. As a result, generic changes have been developed for the standard ITS in NUREG-1431.

3.3 Alternate Source Term

IP3, as a holder of an operating license issued prior to January 10, 1997, is allowed by 10 CFR 50.67, "Accident Source Term," to voluntarily revise its current accident source term used in design basis radiological consequence analyses for a license amendment under 10 CFR Part 50.90. In this license amendment application, the licensee proposed a selective implementation of the AST, as described in RG 1.183 pursuant to 10 CFR 50.67. In general, information provided by RG 1.183 is reflected in Chapter 15.0.1 of the Standard Review Plan (SRP), “Radiological Consequence Analyses Using Alternative Source Term.” The current radiological consequence analysis for the postulated FHA for IP3 is based on the TID 14844 accident source term.

The Commission approved the use of the AST at operating reactors in a Staff Requirement Memorandum (SRM) dated December 8, 1999, stating that, “This action would allow interested licensees to pursue cost-benefit licensing actions to reduce unnecessary regulatory burden without compromising the safety of facility. Many of the alternative source term applications may provide concurrent improvements in overall safety and in reduced occupational exposures.” In a separate SRM dated September 4, 1998, the Commission directed the staff to allow limited or selective application of the AST at operating reactors.

As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criteria of Chapter 15.0.1 of the SRP for the postulated FHA replaces the previous whole body and thyroid dose guidelines of SRP Section 15.7.4 as follows:

	SRP 15.0.1 GDC 19	SRP 15.7.4 GDC 19
Exclusion Area Boundary Low Population Zone	6.3 rem TEDE	75 rem thyroid and 6 rem whole body
Control Room	5 rem TEDE	5 rem whole body, or its equivalent to any part of the body

3.4 Radiological Consequence Analysis

The current radiological consequence analysis for the postulated design basis FHA is based on the accident source term described in TID-14844 and it is provided in the IP3 Final Safety Analysis Report (FSAR) Section 14.2.1. The licensee re-evaluated the radiological consequences of a postulated FHA in the Vapor Containment Building (containment) or in the FSB without taking credit for isolation or filtration in either location. Since the assumptions and parameters used for an FHA inside containment are identical to those for an FHA in the fuel storage building, the resulting radiological consequences are the same regardless of the location of the accident. The licensee selectively implemented the AST, as described in RG 1.183 pursuant to 10 CFR 50.67.

The FHA is postulated to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in a drop of a raised fuel assembly. The licensee assumed that the dropped fuel assembly is damaged to the extent that all fission products in fuel rod gaps in a single fuel assembly are released. Instantaneous release of all noble gases and iodine vapors from the fuel rod gaps from the damaged fuel assembly occurs as gas bubbles up through the water covering the fuel. All noble gases and iodine reaching the containment or the FHB atmosphere are released directly to the environment within 2 hours without filtration.

The licensee concluded in the submittals that the radiological consequences resulting from the postulated FHA in the containment with no credit taken for containment isolation are within the dose acceptance criteria specified in SRP 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” and GDC 19.

The licensee reached this conclusion as a result of:

(1) applying a selective implementation of AST,
(2) taking no credit for containment or FSB isolation,
(3) taking no credit for fission product removal by charcoal adsorbers or high efficiency particulate (HEPA) filters in the FSBEVS,
(4) using an overall decontamination factor of 200 for iodine in elemental and particulate forms in the spent fuel pool water with minimum water depth of 23 feet consistent with the guidelines provided in RG 1.183,
(5) releasing all fission products within 2 hours,
(6) assuming all fuel rods in one fuel assembly with an axial power peaking factor of 1.7 are damaged to the extent that its entire gap activity inventory of the damaged fuel rods is released instantaneously to the surrounding water,
(7) using a fission product decay period of 84 hours (time period from the reactor shutdown to the first fuel movement), and
(8) using the guidance provided in Appendix B to RG 1.183, “Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident.”

In Section 3.2 (Table 3), of RG 1.183, the NRC staff provided the guidelines for non-loss-of-coolant accident (LOCA) fission product inventory fractions in fuel gap. The staff stated in a note to Table 3 that these gap fractions are for the fuel rods that do not exceed the average linear heat generation rate of 6.3 kw/ft and average fuel burnup not exceeding 54 gigawatts per metric ton of uranium (GWD/MTU). The licensee estimated that as many as 10% of the fuel

rods in an assembly could exceed these guidelines. Based on its estimate, the licensee conservatively assumed that 25% of the fuel rods in the damaged fuel assembly would exceed these guidelines. For the fuel rods that exceed the guidelines, the licensee used higher fission product inventory fractions in fuel gap consistent with more conservative gap fractions provided in RG 1.25 , "Assumptions Used for Evaluating the Potential Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," and NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors." For the remaining 75% of the fuel rods that do not exceed the guideline, the licensee used the fission product inventory fractions in fuel gap provided in RG 1.183. The gap fractions used by the licensee are listed in Table 3.

The staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the staff performed an independent confirmatory radiological consequence dose calculation for the postulated FHA. The radiological consequences calculated by the staff are within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1 (6.3 rem TEDE at the exclusion area boundary (EAB)).

Even though the staff performed its confirmatory dose calculations, the staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculations are provided in Table 1, and the major parameters and assumptions used by the licensee are listed in Tables 2 through 4. The radiological consequences at the EAB, at the low-population zone (LPZ), and in the control room as calculated by the licensee are also within the dose criterion specified in GDC 19 and meet the dose acceptance criterion specified in the SRP 15.0.1.

3.5 Control Room Habitability

The licensee normally maintains the IP3 control room at a slightly positive pressure to prevent the introduction of air into the control room from sources other than the 1500 cfm outdoor air makeup flow. The licensee proposed to manually isolate the control room air intakes no later than 20 minutes after the initiation of the postulated LOCA. During this 20-minute period, the licensee assumed an unfiltered air inleakage rate of 1800 cfm. Once the air intakes are isolated, the control room atmosphere would be recirculated through the control room emergency filtration (CREF) system at 1400 cfm with a 400 cfm makeup air. The licensee also assumed 1800 cfm of unfiltered air inleakage to the control room beginning 20 minutes into the accident and continuing throughout the 30-day accident period. The licensee assumed 90% filter efficiencies for removing iodine in elemental and organic forms and for removing fission products in particulate form. The licensee has not performed an integrated control room unfiltered air inleakage test. However, the staff is currently working towards a resolution of generic issues related to control room habitability, with particular focus on the validity of the control room unfiltered air inleakage rates that are commonly assumed in the licensees' analyses of control room habitability. The staff's acceptance of the 1800 cfm unfiltered air inleakage assumption in the license application does not preclude any future generic regulatory action that may become applicable to IP3.

This assessment may be used in subsequent amendments. However, any use of this assessment that involves a relaxation in requirements will require verification (in accordance with the aforementioned resolution of the generic issues related to control room habitability) that the unfiltered inleakage rate is within the limits assumed in the AST assessment. Further, in its March 4 letter, the licensee committed to complete tracer gas testing of the control room envelope to determine the actual flow rate of unfiltered air inleakage. The test will occur on a schedule that will meet the requirements of the NRC's generic issue review.

The results of the licensee's control room radiological consequence calculations are given in Table 1. The major parameters and assumptions used by the staff in its confirmatory dose calculation and by the licensee in its dose calculation are listed in Table 3. The radiological consequences to the control room operator calculated by the licensee and the staff are within the dose criterion specified in 10 CFR 50.67 and, therefore, are acceptable.

3.6 Evaluation of TS Changes

3.6.1 Equipment Hatch Opening, Personnel Airlock Doors, and Penetrations (LCOs 3.9.3.a, 3.9.3.b, and 3.9.3.c)

Current TS LCO 3.9.3.a requires the equipment hatch to be closed or an equipment hatch closure plate, that may include a closed personnel access door, to be installed during Core Alterations or during the movement of irradiated fuel assembles within containment. The licensee proposed to change the requirement that the closure plate access door "be closed" to "capable of being closed."

Current TS LCO 3.9.3.b also requires one door in each personnel airlock to be closed. The licensee proposed to similarly change this to require one door to be capable of being closed.

The licensee proposed to add a Note under LCO 3.9.3.c stating that the penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

In its application, the licensee stated that the proposed TS changes are consistent with the NRC-approved TSTF-68 and TSTF-312, which were approved for use by licensees on August 16, 1999. These two TSTFs were approved by the NRC staff for the use of licensees in proposing changes to plant-specific TSs based on NUREG-1431. TSTF-68 revises LCO 3.9.3 to allow the containment personnel air lock doors to remain open during core alterations or movement of irradiated fuel assemblies within containment. TSTF-312 adds a Note to LCO 3.9.3 to allow containment penetrations that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls during core alterations or movement of irradiated fuel assemblies within containment. The licensee's purpose for proposing to add TSTF-68 and TSTF-312 to the TSs is the same as that approved by the NRC staff in its approval of the TSTFs for NUREG-1431.

In its June 5, 2002, letter, the licensee committed to establish administrative controls to ensure the prompt closure of containment openings in the event of an FHA in the containment building. In addition, the licensee stated that it complies with the 'Reviewers Note' from TSTF-68 regarding the commitment to implement administrative controls for the prompt closure of airlock doors and that the commitment also applies to the equipment hatch closure plate personnel

access door. In its March 4, 2003, letter, the licensee supplemented its description of the administrative controls by stating that the controls assure:

1. appropriate personnel are aware of the open status of the doors and penetration flow paths during movement of irradiated fuel assemblies within containment, and

2. specified individuals are designated and readily available to direct and perform isolation of affected openings in the event of a fuel handling accident, and

3. any obstructions (e.g., cables and hoses) that would prevent rapid closure of an open flow path can be quickly removed. Any cables or hoses to be disconnected should not be supplying services that support personnel safety (e.g., breathing air), and

4. during fuel handling operations and core alterations, ventilation system and radiation monitor availability should be assessed with the goal of minimizing the potential for radioactive releases, following a potential accident, even further below that provided by natural decay that occurs following reactor shutdown.

Based on the description of the administrative controls provided by the licensee for the open air locks, equipment hatch closure plate personnel access door, and containment penetrations with direct access to the outside, the NRC staff concludes that there are sufficient administrative controls to ensure that the air locks and penetrations will be closed following the FHA inside containment. Therefore, the NRC staff finds these changes are consistent with TSTF-68 and TSTF-312 and are acceptable.

In its application, the licensee proposed adding this description to the TS Bases during the implementation of the amendments. Therefore, when the amendments are incorporated into the TSs, the description of the administrative controls will be in the Bases of the TSs. Any changes to the description of the administrative controls will then be controlled by Section 5.5.14 of the Administrative Section of the TSs. This is acceptable to the NRC staff.

3.6.2 Containment Purge System Flow Path and Pressure Relief Line (LCOs 3.9.3.d and LCO 3.9.3.e)

The licensee proposed to delete the current TS LCOs 3.9.3.d and 3.9.3.e regarding the containment purge and containment pressure relief lines that are required to be met if the reactor has been subcritical for less than 550 hours. The licensee also proposed deleting the associated SRs 3.9.3.2 and 3.9.3.4 that would verify the operability. The licensee based its request on the analyses completed to support the adoption of the AST criteria that demonstrates that the dose consequences are acceptable without credit for HEPA and charcoal filtration of the containment ventilation system. The requirement for this filtration during the first 550 hours following shutdown had been determined to be necessary for Vantage+ fuel based on the previous independent dose assessment done in support of Amendment No. 175 dated July 15, 1997.

The NRC staff reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequences resulting from the postulated FHA. The staff's analysis confirmed the licensee's conclusion that the radiological consequences would not exceed the dose criteria specified in 10 CFR 50.67 for control room and the dose acceptance criterion specified in the SRP 15.0.1 for the EAB and LPZ. Therefore, the NRC staff finds the proposed change to be acceptable.

3.6.3 Applicability Statements

In its June 5, 2002, application, the licensee originally proposed to modify the Applicability statement for LCO 3.9.3 to delete "During CORE ALTERATIONS." The licensee stated that the deletion of this applicability was supported by TSTF-51 and that the TSTF further modifies the applicability by defining recently irradiated fuel. In its March 4, 2003, letter, the licensee also revised the remaining portion of the Applicability statement, i.e., "During movement of irradiated assemblies within containment." In this regard, the licensee proposed adding the term "recently" before "irradiated fuel assemblies." The licensee also proposed to remove LCO Action A.1, "Suspend CORE ALTERATIONS," and to change the appropriate TS Bases to define the term "recently irradiated fuel."

In its March 4 letter, the licensee also revised the Applicability statements for LCO 3.3.8 for the FSBEVS Actuation Instrumentation and LCO 3.7.13 for the FSBEVS to also add the term "recently" before "irradiated fuel assemblies." In addition, the LCO Required Action would likewise be modified to add the term "recently" to now require the suspension of movement of recently irradiated fuel in the FSB.

The NRC staff notes that, following a reactor shutdown, the decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed TS changes take advantage of a specific decay period to reduce the radionuclide inventory available for release in the event of an FHA. In the analysis supporting its application, the licensee states that this specific decay period is calculated to be 84 hours. Following the 84-hour decay period, the primary success path for mitigating the FHA no longer includes the operability of the subject engineered safety features (ESFs) components. The FHA is the bounding accident during fuel handling and core alterations. Fuel that has not decayed for 84 hours or longer is termed "recently irradiated fuel" and the subject ESFs must remain operable when moving such fuel.

Applying the "recently irradiated fuel" concept to these TS LCOs provides a mechanism for defining a minimum time for the fission product decay. The decay period of 84 hours has been shown by analysis to provide sufficient decay. Assuming the design basis FHA, the NRC staff ensures that the results of the licensee's analyses of the radiological consequences are within the acceptance criteria of 10 CFR 50.67, "Accident Source Term" and RG 1.183.

The term, "CORE ALTERATION," is defined in the IP3 TS 1.1 as the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. As described in TSTF-51, Revision 2, accidents postulated to occur during core alterations include inadvertent criticality, FHA, and the loading of a fuel assembly or a control component in an incorrect location. Generically, it was concluded that of these off normal occurrences, only the FHA results in cladding damage and potential

radiological release. Consequently, to delete the phrase “during CORE ALTERATIONS” from TS 3.9.3 is consistent with TSTF-51, Revision 2.

3.7 Summary

The NRC staff reviewed the licensee’s analysis and performed a confirmatory assessment of the radiological consequences resulting from the postulated FHA. The doses calculated by the licensee are listed in Table 1. The major parameters and assumptions used by the licensee in its dose calculations and by the staff in its confirmatory dose calculations are listed in Tables 2 through 4. The staff’s analysis confirmed the licensee’s conclusion that the radiological consequences would not exceed the dose criteria specified in 10 CFR 50.67 for control room and the dose acceptance criterion specified in the SRP 15.0.1 for the EAB and LPZ.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 45567). 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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TABLE 1
Radiological Consequences
for
Postulated Design Basis FHA
(rem TEDE)

	EAB	LPZ	Control Room
Fuel-Handling Accident	4.0	1.5	4.8
Dose Criteria	6.7 [(1)]	6.7 [(1)]	5.0 [(2)]

[(1)] From SRP 15.0.1
[(2)] From 10 CFR 50.67

Table 2
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	3085.5 MWt (102% of 3025 MWt)
Radial peaking factor	1.7
Fission product decay period	84 hours
Number of fuel rod damaged	204
Fuel pool water depth	23 ft
Fuel gap fission product inventory for 75% of fuel rods (153 rods)	
Kr-85	10%
I-131	8%
Other iodines and noble gases	5%
Fuel gap fission product inventory for 25% of fuel rods (51 rods)*	
Kr-85	30%
I-131	12%
Other iodines and noble gases	10%
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
Duration of accident	2 hours
Fission product release point	ground level release

* For the fuel rods that exceed average linear heat generation rate of 6.3 kw/ft, together with having a fuel burnup of greater than 54 GWD/MTU.

Table 3
Parameters and Assumptions
Used in
Radiological Consequence Calculations
for Control Room

Parameter	Value
Control room volume	4.72E+4 ft^3
Control room isolation	20 minutes
Control room normal operation	
Unfiltered makeup flow rate	1500 cfm
Unfiltered inleakage rate	1800 cfm
Filtered recirculation flow	0
Control room emergency operation	
Filtered makeup flow rate	400 cfm
Unfiltered inleakage rate	1800 cfm
Filtered recirculation flow	1000 cfm
CREF system filter efficiencies	
Elemental iodine	90%
Organic iodine	90%
Aerosol (particulate)	90%

TABLE 4
Meteorological Data [(1)]

Exclusion Area Boundary

Time	X/Q (sec/m^3)
0 - 2 hrs	1.03E-03

Low Population Zone

Time	X/Q (sec/m^3)
0 - 2 hrs	3.8E-04
2 - 4 hrs	1.8E-04
4 - 8 hrs	1.8E-04
8 - 24 hrs	1.9E-05
24 - 270 hrs	1.7E-06

Control Room

Time	X/Q (sec/m^3)
0 - 2 hrs	2.30E-03
2 - 24 hrs	1.15E-03
1 - 30 days	5.50E-04

[(1)] Original licensing basis in IP3 FSAR Chapter 14.3