

April 28, 2005

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: PILGRIM NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE:
ALTERNATIVE SOURCE TERM FOR THE FUEL HANDLING ACCIDENT
DOSE CONSEQUENCES (TAC NO. MC2705)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 215 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated April 14, 2004, as supplemented by letter dated December 15, 2004.

This amendment eliminates secondary containment operability requirements when handling sufficiently decayed irradiated fuel or performing core alterations. The secondary containment is still required to be operable during operations with the potential to drain the reactor vessel.

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/

John P. Boska, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 215 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

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ENTERGY NUCLEAR GENERATION COMPANY
ENTERGY NUCLEAR OPERATIONS, INC.
DOCKET NO. 50-293
PILGRIM NUCLEAR POWER STATION
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 215
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by Entergy Nuclear Operations, Inc. (the licensee) dated April 14, 2004, as supplemented on December 15, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;
 - 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 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, 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 issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 28, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 215

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

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

3/4.2-24
3/4.7-11
3/4.7-12
3/4.7-13
3/4.7-14
3/4.7-15
3/4.7-16
B3/4.7-10
B3/4.7-11
B3/4.7-12
B3/4.7-13
B3/4.7-14

Insert

3/4.2-24
3/4.7-11
3/4.7-12
3/4.7-13
3/4.7-14
3/4.7-15
3/4.7-16
B3/4.7-10
B3/4.7-11
B3/4.7-12
B3/4.7-13
B3/4.7-14

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 215 TO FACILITY OPERATING LICENSE NO. DPR-35

ENERGY NUCLEAR GENERATION COMPANY

ENERGY NUCLEAR OPERATIONS, INC.

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated April 14, 2004, as supplemented by letter dated December 15, 2004, Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request for changes to the Pilgrim Nuclear Power Station (PNPS) Technical Specifications (TSs) to the Nuclear Regulatory Commission (NRC or the Commission). The supplement dated December 15, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 12, 2004 (69 FR 60679).

The proposed change revises the PNPS TSs to implement Technical Specifications Task Force Traveler 51 (TSTF-51), "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations." The licensee also proposed to selectively implement an alternative source term (AST) per Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67 to perform the radiological consequences analysis of the design-basis fuel handling accident (FHA) which supports the proposed TS changes.

The proposed changes would allow the movement of fuel that has not been recently irradiated within the containment without requiring secondary containment, standby gas treatment system (SGTS), or control room (CR) high efficiency air filtration system (CRHEAFS) operability. "Recently irradiated" fuel is defined in TSTF-51 as fuel that has not been part of a critical core within a time period that is shown through analyses to allow for enough radiological decay so that the radiological consequences of a design-basis FHA will remain acceptable.

For adoption of TSTF-51, the licensee requested changes to the following TSs:

- 3/4.2.D, Table 3.2.D, "Radiation Monitoring Systems That Initiate and/or Isolate"
- 3/4.7.B.1, "Standby Gas Treatment System"
- 3/4.7.B.2, "Control Room High Efficiency Air Filtration System"
- 3/4.7.C. "Secondary Containment"

2.0 REGULATORY EVALUATION

The construction permit for PNPS was issued by the Atomic Energy Commission (AEC) on August 26, 1968, a low-power license was issued on June 8, 1972, and a full-power license was issued on September 15, 1972. The plant design approval for the construction phase was based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as "draft GDC"). The AEC published the final rule that added Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "final GDC").

Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. In accordance with a staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (Agencywide Documents and Management System (ADAMS) Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971, which included PNPS. The PNPS Updated Final Safety Analysis Report (UFSAR), Appendix F, provides an evaluation of the design bases of PNPS against the draft GDC.

Although the original approval basis for PNPS was the draft GDC, the licensees for PNPS have made changes to the facility over the life of the plant that may have invoked some of the final GDC. The extent to which the final GDC have been invoked can be found in specific sections of the UFSAR and in other PNPS design and licensing basis documentation. For convenience, the licensee and the NRC staff usually refer to the final GDC rather than the draft GDC when discussing licensing actions.

This safety evaluation (SE) input addresses the impact of the proposed changes on previously-analyzed design basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50 Appendix A, GDC 19, "Control Room," as supplemented by Section 6.4 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in SRP Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," in performing this review. The NRC staff also considered relevant information in the PNPS UFSAR, TSs, and TSTF-51.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of DBAs, performed by Entergy in support of its proposed license amendment. Information regarding these analyses was provided in Section 4.0 and Attachments 2 and 3 of the submittal dated April 14, 2004, and in a supplementary letter dated December 15, 2004. The NRC staff reviewed the assumptions, inputs, and methods used by Entergy to assess these impacts. The NRC staff performed independent calculations to confirm the conservatism of the licensee's analyses. However, the findings of this SE are based on the descriptions of the licensee's analyses and other supporting information docketed by Entergy.

TSTF-51, Revision 2 allows for removal of the TS requirements for engineered safeguards features, such as secondary containment and standby gas treatment (SGT), to be operable during movement of fuel, once sufficient radioactive decay has taken place to ensure that offsite doses remain well within (i.e., 25 percent of) 10 CFR Part 100 limits. TSTF-51 placed the following reviewer's note in the basis for the standard TS 3.6.4.1, "Secondary Containment":

The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by analysis that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10 CFR 100).

Additionally, licensees adding the term "recently" must make the following commitment which is consistent with draft NUMARC 93-01, Revision 3, Section 11.2.6, "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions", subheading "Containment - Primary (PWR [pressurized-water reactor])/Secondary (BWR [boiling-water reactor])".

"The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel:

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.

The purpose of the "prompt methods" mentioned above are 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."

When draft NUMARC 93-01, Revision 3, was issued in the final version, Section 11.2.6 was renumbered to Section 11.3.6 and the guidelines quoted above were numbered 11.3.6.5 and modified slightly as follows:

"...the following guidelines should be included in the assessment of systems removed from service:

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is 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 NRC staff considers the final version and the draft version of this section of NUMARC 93-01, Revision 3, to be functionally equivalent. In attachment six to the licensee's submittal of April 14, 2004, the licensee committed to implement the guidelines of section 11.3.6.5 of NUMARC 93-01, Revision 3, prior to the implementation of this license amendment. The NRC staff finds this meets the commitment required by TSTF-51, Revision 2.

Because the licensee has applied to implement an AST selectively for the FHA, the regulatory dose criteria of 10 CFR 50.67 are used in lieu of the 10 CFR Part 100 dose limits in determining acceptability of the changes.

3.1 Technical Specification Changes

The licensee has proposed to make the following changes to the PNPS TSs to change the operability requirements of certain engineered safeguards feature (ESF) systems during refueling operations. These changes are based on the application of the TSTF-51 program which has been previously accepted by the NRC. In general, the TSTF-51 program allows certain systems to be in a non-operational status once a period of initial fuel decay has occurred. Fuel in the period prior to the completion of this decay is referred to as "recently irradiated fuel." The time period of the decay is fuel-cycle specific, is calculated, and is defined in the technical basis document for each fuel cycle. The period of time for "recently irradiated fuel" for the current fuel cycle is 24 hours.

As required by the TSTF-51 program, the licensee has committed to the shutdown provisions of NUMARC 93-01. The guidance in NUMARC 93-01 states that for the period of time that ESF systems are not operable, provisions will be made in the event of an accidental release to isolate containment and to process the release using appropriate filter systems to reduce the dose even lower than those which are required by the regulations.

The following TS changes were reviewed by the staff:

a. **TS Table 3.2.D (TS Page 3/4.2-24), "Radiation Monitoring Systems that Initiate and/or Isolate":**

The "Action" statements A and B are revised as follows:

Current TS	Proposed TS
A. Cease operation of the refueling equipment. B. Isolate secondary containment and start the standby gas treatment system.	A. Cease operation of the refueling equipment. <i>movement of recently irradiated fuel assemblies and operations with potential to drain the reactor vessel (OPDRVs).</i> B. Isolate secondary containment and start the standby gas treatment system <i>during movement of recently irradiated fuel assemblies and OPDRVs</i>

The proposed change permits the continued operation of refueling equipment and removes the requirement for isolation of the secondary containment and the starting of the standby gas treatment system after the time period defined as "recently" has passed. The staff finds this change to be consistent with the TSTF-51 program. The licensee's analysis of the FHA complies with the dose criteria in 10 CFR 50.67 and the regulatory dose acceptance criteria of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and takes no credit for containment closure or the operation of the SGT system. Therefore, the NRC staff finds this change to be acceptable.

b. TS 3/4.7.B.1 .a, .c, and .e (TS pages 3/4.7-11, 12, and 13) for Standby Gas Treatment System and Control Room High Efficiency Air Filtration System.

(l) TS 3/4.7.B.1.a is revised as follows:

Current TS	Proposed TS
<p>1. Standby Gas Treatment System</p> <p>a. Except as specified in 3.7.B.1.c or 3.7.B.1.e below, both trains of the standby gas treatment shall be operable when in the Run, Startup, and Hot Shutdown MODES, during movement of irradiated fuel assemblies in the secondary containment, and during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and during operations with a potential for draining the reactor vessel (OPDRVs),</p>	<p>1. Standby Gas Treatment System</p> <p>a. Except as specified in 3.7.B.1.c or 3.7.B.1.e below, both trains of the standby gas treatment shall be operable when in the Run, Startup, and Hot Shutdown MODES, during movement of <i>recently</i> irradiated fuel assemblies in the secondary containment, and during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS and during operations with a potential for draining the reactor vessel (OPDRVs),</p>

The proposed change removes operability requirements on the standby gas treatment system after the time period defined as “recently” has passed and during movement of new fuel over the spent fuel pool (SFP) and during CORE ALTERATIONS. The licensee has shown that accidents which could occur during the movement of new fuel over the SFP or during CORE ALTERATIONS are bounded by the FHA results and the NRC staff concurs with the licensee's conclusion. Dropping a new fuel bundle is less severe than dropping an irradiated fuel bundle, as the same number of fuel rods are postulated to be damaged but the fuel rods in the new fuel bundle have a much lower inventory of radioactive isotopes than irradiated fuel rods. CORE ALTERATIONS are defined in the PNPS TS 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..." The accidents postulated to occur during CORE ALTERATIONS, in addition to the fuel handling accident, are inadvertent criticality due to a control rod removal error, and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage, and are therefore bounded by the design basis FHA. The staff finds this TS change to be consistent with the TSTF-51 program. The licensee's analysis of the FHA complies with the dose criteria in 10 CFR 50.67 and the regulatory dose acceptance criteria of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and takes no credit for operation of the SGT system. Therefore, the NRC staff finds this change to be acceptable.

(ii) TS 3/4.7.B.1.c is revised as follows:

Current TS	Proposed TS
<p>c. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, continued reactor operation, irradiated fuel handling, or new fuel handling over the spent fuel pool is permissible only during the succeeding seven days providing that within 2 hours all active components of the other standby gas treatment train are verified to be operable and the diesel generator associated with the operable train is operable.</p> <p>If the system is not made fully operable within 7 days, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within the next 36 hours and fuel handling operations shall be terminated within 2 hours.</p> <p>Fuel handling operations in progress may be completed.</p>	<p>c. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, continued reactor operation, irradiated fuel handling, or new fuel handling over the spent fuel pool is permissible only during the succeeding seven days providing that within 2 hours all active components of the other standby gas treatment train are verified to be operable and the diesel generator associated with the operable train is operable.</p> <p>If the system is not made fully operable within 7 days, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within the next 36 hours. and fuel handling operations shall be terminated within 2 hours.</p> <p>Fuel handling operations in progress may be completed.</p>

The proposed change removes the restrictions on irradiated fuel handling, or new fuel handling over the SFP when one SGT train is not operable. It maintains all the restrictions of reactor operation when one SGT train is not operable. The staff finds this change to be consistent with the TSTF-51 program. The licensee's analysis of the FHA complies with the dose criteria in 10 CFR 50.67 and the regulatory dose acceptance criteria of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and takes no credit for operation of the SGT system. Therefore, the NRC staff finds this change to be acceptable.

(iii) TS 3/4.7.B.1.e is revised as follows:

Current TS	Proposed TS
<p>e. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason during Refuel Outages, refueling operations are permissible only during the succeeding 7 days providing that within 2 hours all active components of the other train are verified to be operable and the diesel generator associated with the operable train is operable.</p> <p>If the system is not made fully operable within 7 days,</p> <p>l) place the operable train in operation immediately</p> <p><u>or</u></p> <p>ii) suspend movement of irradiated fuel assemblies in secondary containment or new fuel handling over the spent fuel pool or core.</p> <p>Any fuel assembly movement in progress may be completed.</p>	<p>e. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, during Refuel Outages, refueling operations <i>movement of recently irradiated fuel assemblies and operations with a potential for draining the reactor vessel (OPDRVs)</i> are permissible only during the succeeding 7 days providing that within 2 hours all active components of the other train are verified to be operable and the diesel generator associated with the operable train is operable.</p> <p>If the system is not made fully operable within 7 days,</p> <p>l) place the operable train in operation immediately.</p> <p><u>or</u></p> <p>ii) suspend movement of <i>recently</i> irradiated fuel assemblies in secondary containment <i>and initiate actions to suspend OPDRVs. or new fuel handling over the spent fuel pool and core.</i></p> <p>Any fuel assembly movement in progress may be completed.</p>

The proposed change alters the applicability of the TS from all refueling operations to only those refueling operations that are related to movement of recently irradiated fuel assemblies and OPDRVs. Activities that pertain to fuel that has decayed past the time defined as “recently” and new fuel handling over the SFP and core would be permitted since the potential accidental release would be bounded by the FHA analysis. The staff finds this change to be consistent with the TSTF-51 program. The licensee's analysis of the FHA complies with the dose criteria in 10 CFR 50.67 and the regulatory dose acceptance criteria of RG 1.183, "Alternative

Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and takes no credit for operation of the SGT system. Therefore, the NRC staff finds this change to be acceptable.

c. TS 3/4.7.B.2.a (TS pages 3/4.7-14) for Control Room High Efficiency Air Filtration System.

(l) TS 3/4.7.B.2.a is revised as follows:

Current TS	Proposed TS
<p>a. Except as specified in Specification 3.7.B.2.c or 3.7.B.2.e below, both trains of the Control Room High Efficiency Air Filtration System used for the processing of inlet air to the control room under accident conditions shall be operable when in the Run, Startup, and Hot Shutdown MODES, during movement of irradiated fuel assemblies in the secondary containment, and during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and during operations with a potential for draining the reactor vessel (OPDRVs),</p> <p><u>or</u></p> <p>the reactor shall be in cold shutdown within the next 36 hours.</p>	<p>a. Except as specified in Specification 3.7.B.2.c or 3.7.B.2.e below, both trains of the Control Room High Efficiency Air Filtration System used for the processing of inlet air to the control room under accident conditions shall be operable when in the Run, Startup, and Hot Shutdown MODES, during movement of <i>recently</i> irradiated fuel assemblies in the secondary containment, and during the movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and during operations with a potential for draining the reactor vessel (OPDRVs),</p> <p><u>or</u></p> <p>the reactor shall be in cold shutdown within the next 36 hours.</p>

The proposed change alters the applicability of the TS from all refueling operations to only those refueling operations that are related to movement of recently irradiated fuel assemblies. Activities that pertain to fuel that has decayed past the time defined as "recently," new fuel handling over the SFP and core, and CORE ALTERATIONS would be permitted since the potential accidental release would be bounded by the FHA analysis. The staff finds this change to be consistent with the TSTF-51 program. The licensee's analysis of the FHA complies with the dose criteria in 10 CFR 50.67 and the regulatory dose acceptance criteria of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and takes no credit for operation of the CRHEAFS. Therefore, the NRC staff finds this change to be acceptable.

(ii) TS 3/4.7.B.2.c is revised as follows:

Current TS	Proposed TS
<p>c. From and after the date that one train of the Control Room High Efficiency Air Filtration System is made or found to be inoperable for any reason, reactor operation, irradiated fuel handling, or new fuel handling over the spent fuel pool is permissible only during the succeeding 7 days providing that within 2 hours all active components of the other CRHEAF train are verified to be operable and the diesel generator associated with the operable train is operable. If the system is not made fully operable within 7 days, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within the next 36 hours and fuel handling operations shall be terminated within 2 hours. Fuel handling operations in progress may be completed.</p>	<p>c. From and after the date that one train of the Control Room High Efficiency Air Filtration System is made or found to be inoperable for any reason, reactor operation irradiated fuel handling, or new fuel handling over the spent fuel pool is permissible only during the succeeding 7 days providing that within 2 hours all active components of the other CRHEAF train are verified to be operable and the diesel generator associated with the operable train is operable. If the system is not made fully operable within 7 days, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within the next 36 hours. and fuel handling operations shall be terminated within 2 hours. Fuel handling operations in progress may be completed.</p>

The proposed change removes the restrictions on irradiated fuel handling, or new fuel handling over the SFP when one CRHEAF train is not operable. It maintains all the restrictions of reactor operation when one CRHEAF train is not operable. The staff finds this change to be consistent with the TSTF-51 program. The licensee's analysis of the FHA complies with the dose criteria in 10 CFR 50.67 and the regulatory dose acceptance criteria of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and takes no credit for operation of the CRHEAFS. Therefore, the NRC staff finds this change to be acceptable.

(iii) TS 3/4.7.B.2.e is revised as follows:

Current TS	Proposed TS
<p>e. From and after the date that one train of the Control Room High Efficiency Air Filtration System is made or found to be inoperable for any reason during Refuel Outages, refueling operations are permissible only during the succeeding 7 days providing that within 2 hours all active components of the other train are verified to be operable and the diesel generator associated with the operable train is operable.</p> <p>If the system is not made fully operable within 7 days,</p> <p>l) perform surveillance 4.7.B.2.b.4 for the operable CRHEAF every 24 hours</p> <p><u>or</u></p> <p>ii) suspend movement of irradiated fuel assemblies in secondary containment or new fuel handling over the spent fuel pool or core.</p> <p>Any fuel assembly movement in progress may be completed.</p>	<p>e. From and after the date that one train of the Control Room High Efficiency Air Filtration System is made or found to be inoperable for any reason during Refueling Outages, refueling operations <i>movement of recently irradiated fuel assemblies and operations with a potential for draining the reactor vessel (OPDRVs)</i> are permissible only during the succeeding 7 days providing that within 2 hours all active components of the other train are verified to be operable and the diesel generator associated with the operable train is operable.</p> <p>If the system is not made fully operable within 7 days,</p> <p>l) perform surveillance 4.7.B.2.b.4 for the operable CRHEAF every 24 hours.</p> <p><u>or</u></p> <p>ii) suspend movement of <i>recently</i> irradiated fuel assemblies in secondary containment <i>and initiate actions to suspend OPDRVs or new fuel handling over the spent fuel pool or core.</i> Any fuel assembly movement in progress may be completed.</p>

The proposed change alters the applicability of the TS from all refueling operations to only those refueling operations that are related to movement of recently irradiated fuel assemblies and OPDRVs. Activities that pertain to fuel that has decayed past the time defined as “recently” and new fuel handling over the SFP and core would be permitted since the potential accidental release would be bounded by the FHA analysis. The staff finds this change to be consistent with the TSTF-51 program. The licensee’s analysis of the FHA complies with the dose criteria in 10 CFR 50.67 and the regulatory dose acceptance criteria of RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” and takes no credit for operation of the CRHEAFS. Therefore, the NRC staff finds this change to be acceptable.

d. TS 3/4.7.C.1 and .2 Secondary Containment are revised as follows:

Current TS	Proposed TS
<p>1. Secondary containment shall be OPERABLE when in the Run, Startup and Hot Shutdown MODES, during movement of irradiated fuel assemblies in the secondary containment, and during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and during operations with a potential for draining the reactor vessel (OPDRVs).</p> <p>2. a. With Secondary Containment inoperable when in the Run, Startup and Hot Shutdown MODES, restore Secondary Containment to OPERABLE status within 4 hours.</p> <p>b. Required Action and Completion Time of 2.a not met, be in HOT Shutdown in 12 hours AND Cold Shutdown within 36 hours.</p> <p>c. With Secondary Containment inoperable during movement of irradiated fuel assemblies in the secondary containment, and during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and during OPDRVs, immediately</p> <ol style="list-style-type: none"> 1. Suspend movement of irradiated fuel assemblies in the secondary containment. <u>AND</u> 2. Suspend movement of new fuel over the spent fuel pool. <u>AND</u> 3. Suspend CORE ALTERATIONS <u>AND</u> 4. Initiate action to suspend OPDRVs. 	<p>1. Secondary containment shall be OPERABLE when in the Run, Startup and Hot Shutdown MODES, during movement of <i>recently</i> irradiated fuel assemblies in the secondary containment, and during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and during operations with a potential for draining the reactor vessel (OPDRVs).</p> <p>2. a. With Secondary Containment inoperable when in the Run, Startup and Hot Shutdown MODES, restore Secondary Containment to OPERABLE status within 4 hours.</p> <p>b. Required Action and Completion Time of 2.a not met, be in HOT Shutdown in 12 hours AND Cold Shutdown within 36 hours.</p> <p>c. With Secondary Containment inoperable during movement of <i>recently</i> irradiated fuel assemblies, and during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and during OPDRVs, immediately:</p> <ol style="list-style-type: none"> 1. Suspend movement of <i>recently</i> irradiated fuel assemblies in the secondary containment. <u>AND</u> 2. Suspend movement of new fuel over the spent fuel pool. <u>AND</u> 3. Suspend CORE ALTERATIONS <u>AND</u> 24. Initiate action to suspend OPDRVs.

The proposed TS change would permit the secondary containment to be in the not operable condition for refueling activities involving fuel that had decayed past the time defined as “recently.” The licensee has committed to NUMARC 93-01 which requires the secondary containment to be isolated in the event of an accidental release to further reduce the dose even lower than that required by the regulations. The staff finds this change to be consistent with the TSTF-51 program. The licensee has performed an analysis that demonstrates that the release associated with an FHA would not exceed the regulatory guidelines contained in 10 CFR 50.67 and RG 1.183, and the NRC staff concurs with the results of this analysis and finds this TS change to be acceptable.

The licensee stated in the submittal that they may consider making a temporary penetration to the PNPS secondary containment during the refueling outage to provide additional flexibility in moving equipment. The staff has not reviewed the addition of an additional penetration. The TSTF-51 program referenced by the licensee does not make any reference to the addition of additional penetrations. This SE does not make a finding with respect to acceptability of any additional temporary penetrations. If the licensee pursues the concept of an additional penetration, it is expected that the licensee would perform the appropriate safety analysis and because of the licensee’s commitment to NUMARC 93-01, provision would be made to rapidly close off this penetration in the event of an accidental release during refueling.

3.2 Radiological Consequences of the FHA

In the postulated FHA, a fuel assembly is assumed to be dropped, thereby damaging 151 fuel rods during fuel handling. To support the proposed TS changes, the licensee does not take credit for secondary containment isolation or filtration by the SGT system or CRHEAFS in its analysis, and assumed the FHA occurred 24 hours after reactor shutdown from full power.

The entire gap activity from the damaged fuel is assumed to be released directly to the outside atmosphere over a 2-hour period. The licensee calculated the activity in the gap of the fuel rods assuming the assembly has been operated at 100.5 percent of the maximum core thermal power times a maximum radial peaking factor of 2.1. The analysis assumed the RG 1.183 Table 3 non-loss-of-coolant-accident gap fractions. In accordance with RG 1.183, the licensee assumed the iodine species released from the fuel gap to the water was 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic and the effective iodine decontamination factor for the water pool was 200. The iodine species above the pool were assumed to be 57 percent elemental and 43 percent organic, in accordance with RG 1.183.

There are two possible release points for the FHA when the reactor building is open to the environment and the SGT system and secondary containment are inoperable. The release may occur from the reactor building vent or through the open reactor building truck airlock door. The licensee assumed a ground level release from the reactor building vent, which they determined to be the bounding release point. Entergy calculated new values for both offsite and control room atmospheric dispersion factors for use in the DBA dose calculation. In Table 4 of Attachment 1 to the December 15, 2004 submittal, Entergy reported doses at the exclusion area boundary (EAB) and low-population zone (LPZ) which were calculated using gamma atmospheric dispersion factors as opposed to the usual concentration factors for the offsite receptors. In Attachment 2 to the December 15, 2004 submittal, the licensee also calculated the offsite doses using the concentration atmospheric dispersion factors. The staff does not find the licensee’s use of gamma atmospheric dispersion factors to be appropriate or

acceptable. The difference in the licensee's calculated dose when using the concentration atmospheric dispersion factors is small, as documented in Table A.1 of Attachment 2 to the December 15, 2004 submittal. The staff bases its finding of acceptability on the licensee's calculation using the concentration atmospheric dispersion factors. Discussion of the NRC staff's review of the licensee's revised atmospheric dispersion factors is below.

To calculate the dose in the CR, the licensee assumed a high value (9000 cfm) of unfiltered flow into the CR in order to bound the normal fresh air intake flow rate of 7200 cfm. This equates to 1800 cfm of unfiltered CR envelope leakage. To demonstrate the impact of the CR unfiltered intake assumption on the calculated dose in the CR, Entergy performed a sensitivity case assuming a lower bounding value for the CR fresh air intake flow rate of 1000 cfm. All other assumptions were the same as in the analysis described above. The decrease in calculated dose was 0.017 rem for the 1000 cfm intake case as compared to the 9000 cfm intake case. This result indicates that the impact of the assumed value for intake flow rate is small when the CR ventilation system is in the normal mode of operation for the FHA .

The staff reviewed the information provided in the licensee's submittal, as supplemented, and also performed an independent calculation that confirmed the licensee's dose results. Entergy's analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 1 of this SE. The licensee's calculated dose results are given in Table 2. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for an FHA. These total effective dose equivalent (TEDE) criteria are 6.3 rem at the EAB for the worst two hours, 6.3 rem at the LPZ for the duration of the accident and 5 rem in the CR for the duration of the accident.

3.3 Atmospheric Dispersion Estimates

Entergy used onsite hourly meteorological data collected during calendar years 1996-2000 to generate new atmospheric dispersion factors (χ/Q values) for use in this License Amendment Request (LAR). These data were provided for staff review in the form of hourly meteorological data files (for input into the ARCON96 atmospheric dispersion computer code) consisting of hourly wind data from the 10-meter and 67.1-meter levels on the onsite meteorological tower. A joint wind speed, wind direction and atmospheric stability frequency distribution (for input to the PAVAN atmospheric dispersion computer code) was also provided and compiled with wind data from the 10-meter level. Stability class was calculated using the temperature difference between the 67.1-meter and 10-meter levels for both sets of data. Entergy stated that the onsite meteorological monitoring system used to collect these data met the RG 1.23, "Onsite Meteorological Programs," siting criteria. The data were used to generate CR, EAB, and LPZ χ/Q values for the FHA evaluated in this LAR. All releases were assumed to be ground level. The resulting atmospheric dispersion factors represent a change from those used in the current PNPS UFSAR Chapter 14 accident analysis.

The staff performed a quality review of the ARCON96 hourly meteorological database 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. Examination of the data revealed that stable and neutral atmospheric conditions were generally reported to occur at night and unstable and neutral conditions were generally reported to occur during the day, as expected. Wind speed, wind direction, and stability class

frequency distributions for each measurement channel were reasonably similar from year to year. A comparison of joint frequency distribution derived by the staff from the ARCON96 hourly data with the joint frequency distribution developed by Entergy showed reasonably good agreement.

In summary, the staff reviewed the available information relative to the onsite meteorological measurements program and the resulting ARCON96 and PAVAN meteorological data input files provided by Entergy. On the basis of this review, the staff concludes that these data provide an acceptable basis for making estimates of atmospheric dispersion for DBA assessments.

Entergy used guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," and the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") to generate the CR atmospheric dispersion factors. The data recovery rate for the period-of-record provided as input to ARCON96 exceeded 95 percent. Two potential release pathways (i.e., the reactor building vent and truck lock door) were modeled as ground-level point sources with the difference in heights between the release point and receptor taken into consideration. Entergy determined that a postulated release from the reactor building vent would be the more limiting case and used those χ/Q values in its dose assessment. The NRC staff qualitatively reviewed the inputs to the ARCON96 computer runs and found them generally consistent with site configuration drawings and NRC staff practice. The NRC staff also made an independent evaluation of the resulting atmospheric dispersion estimates by running the ARCON96 computer model and obtained results similar to those calculated by Entergy.

In summary, the staff reviewed Entergy's assessments of CR post-accident dispersion conditions generated from their meteorological data and atmospheric dispersion modeling. The resulting CR χ/Q values are presented in Table 1. On the basis of this review, the staff concludes that these χ/Q values are acceptable for use in the FHA CR dose assessment.

Entergy calculated two sets of EAB and LPZ χ/Q values using the AEOLUS-3 atmospheric dispersion computer code, one set referred to as "concentration χ/Q values" and the second set as "gamma χ/Q values." Entergy stated that this computer code implements guidance in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." However, RG 1.145 does not address gamma χ/Q values. Entergy's concentration χ/Q values are those of the type generally applied to ground-level release EAB and LPZ dose assessments and, in this case, are more limiting than the gamma χ/Q values. Therefore, as a matter of expediency, due to the small difference in resultant doses, and the fact that use of the gamma χ/Q values would result in a lower (i.e., less limiting) dose in this assessment, the NRC staff did not review Entergy's gamma χ/Q values as part of this LAR. The NRC staff also did not review Entergy's atmospheric dispersion code, but did qualitatively review inputs to the computer runs as applied to calculation of the concentration χ/Q values. Entergy considered all releases to be ground level and assumed an adjacent building height of 43.6 m and minimum cross-sectional area of 1886 m². The NRC staff found the inputs for calculation of the concentration χ/Q values consistent with information in the PNPS UFSAR and NRC staff practice. The staff also made an independent evaluation of the atmospheric dispersion estimates by running the PAVAN computer model (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations") which implements RG 1.145 and obtained

results similar to Entergy's concentration χ/Q values. The NRC staff therefore has concluded that Entergy's concentration χ/Q values are acceptable for use in this application.

In summary, the staff reviewed Entergy's assessments of EAB and LPZ post-accident dispersion conditions generated from their meteorological data and atmospheric dispersion modeling for the concentration χ/Q values. The resulting EAB and LPZ concentration χ/Q values are presented in Table 1. On the basis of this review, the staff concludes that these χ/Q values are acceptable for use in the FHA EAB and LPZ dose assessments.

3.4 Control Room Habitability

On June 12, 2003, the staff issued Generic Letter (GL) 2003-01, "Control Room Habitability." This GL identifies staff concerns regarding the reliability of current surveillance testing to identify and quantify CR inleakage, and requests licensees to confirm the most limiting unfiltered inleakage into their CR envelope. Entergy was required by the GL to respond to the information request within 180 days of its issue. The PNPS 60-day response was submitted to the NRC by letter dated August 6, 2003, and stated that they could not meet the 180-day schedule to respond to the GL and proposed an alternative course of action. The staff has determined that there is reasonable assurance that the PNPS CR will be habitable during an FHA with the proposed changes to containment closure TSs, and this amendment may be approved prior to the staff's review of the PNPS response to the GL. The staff bases this determination on the bounding dose analyses provided by the licensee and the verification of the CR unfiltered inleakage assumption through tracer gas testing, planned for November 2005. The staff's approval of this amendment does not relieve Entergy of addressing the information requests in GL 2003-01 and does not imply that the staff would necessarily find the analysis in this amendment acceptable as a response to information request 1(a) in GL 2003-01.

3.5 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by Entergy to assess the radiological impacts of implementing TSTF-51 and selectively implementing an AST for the FHA at PNPS. The staff finds that Entergy used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by Entergy to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will continue to comply with these criteria. Therefore, the proposed changes are acceptable with regard to the radiological consequences of postulated DBAs.

This licensing action is considered a selective implementation of the AST. With this approval, the selected characteristics of the AST and the TEDE criteria become the design basis for the analysis of the FHA at PNPS. This approval is limited to this specific implementation. Subsequent modifications based on the selected characteristics of the AST incorporated into the PNPS design basis by this action may be possible under the provisions of 10 CFR 50.59. However, the selected characteristics of the AST, as described above, and the TEDE criteria may not be extended to other aspects of the plant design or operation without prior NRC review under 10 CFR 50.67. All future radiological analyses performed to demonstrate compliance with regulatory requirements which are within the scope of the selective implementation shall

address the selected characteristics of the AST and the TEDE criteria as described in the PNPS design basis.

**Table 1
FHA Analysis Assumptions**

Reactor power	2038 megawatts-thermal		
Radial peaking factor	2.1		
Fission product decay period	24 hours		
Number of fuel rods damaged	151		
Fuel gap fission product inventory			
I-131	8 percent		
Kr-85	10		
Other iodines and noble gases	5		
Fuel pool water depth	23 feet		
Pool iodine effective decontamination factor	200		
Chemical form of iodine above pool			
Elemental	57 percent		
Organic	43 percent		
Duration of release	2 hours		
Control room volume	34,280 ft ³		
Normal ventilation unfiltered intake	9000 cfm		
Atmospheric dispersion factors, sec/m ³			
Ground level release from reactor building vent			
<u>Period</u>	<u>EAB</u>	<u>LPZ</u>	<u>Control Room</u>
0-2 hours	7.479E-04	3.692E-05	1.76E-03
2-8 hours		1.915E-05	1.25E-03
8-24 hours		1.066E-05	4.26E-04
24-96 hours		4.339E-06	3.67E-04
96-720 hours		1.194E-06	3.15E-04

Table 2
Licensee Calculated Radiological Consequences

<u>FHA</u>	<u>EAB</u>	<u>TEDE (rem)</u>	
		<u>LPZ</u>	<u>Control Room</u>
9000 cfm control room air intake	1.767	0.0933	2.863
Dose acceptance criteria	6.3	6.3	5

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts 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 (69 FR 60679). 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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