Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

Michael A. Balduzzi
Site Vice President

December 15, 2004

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

SUBJECT: Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Proposed License Amendment for a Limited Scope Application of the Alternate Source Term (NUREG-1465) for Re-evaluation of the Fuel Handling Accident Dose Consequences, Rev. 1(TAC NO. MC2705)

REFERENCE:

1. Entergy Letter, 02.04.003, Proposed License Amendment for a Limited Scope Application of the Alternate Source Term (NUREG-1465) for Re-evaluation of the Fuel Handling Accident Dose Consequences, dated, April 14, 2004.

2. NRC Request for Additional Information, dated October 13, 2004

LETTER NUMBER: 2.04.115

Dear Sir:

By this letter, Entergy submits a revision to the proposed license amendment that was submitted by Reference 1 to change the requirements associated with handling irradiated fuel and performing core alterations. These attachments are revised to reflect updated fuel handling calculations and responses to an NRC request for additional information (Reference 2).

The revised submittal does not change the no significant hazards consideration determination previously submitted by Reference 1.

The scope and content of the proposed changes is similar to the recently NRC approved Technical Specification changes for James A. FitzPatrick (TAC No. MB5328) and Duane Arnold (TAC NO. MB1569).

The commitments made in this letter by the licensee are listed in Attachment 6 of this letter.

Entergy requests NRC review and approval of this proposed change by March 1, 2005 to support the Pilgrim refueling outage-15 in April 2005.

Please contact Mr. Bryan Ford at (508) 830-8403, if you have any questions.

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

Executed on the 15th day of December 2004.

Sincerely,

Michael A. Balduzzi

Michael A. Balduzzi

WGL/dm

Attachments:

1. Proposed License Amendment for a Limited Scope Application of the Alternate Source Term Guidelines in NUREG-1465 for Re-evaluation of the Fuel Handling Accident Dose Consequences (30 pages), Rev. 1
2. Areva Document No. 32-5052589-01, "Radiological Consequences of a Design-Basis Fuel Handling Accident Based on the Alternate Source Term Methodology", (141 pages)
3. Areva Document No. 32-5052821-01, "Determination of Atmospheric Dispersion Factors for Accident Analyses Using Reg Guide 1.145 and 1.194 Methodologies" (80 pages); Areva Document No. 32-5052036-00, "Evaluation of Pilgrim Nuclear Power Station 1996-2001 Meteorological Data" (32 pages); and Areva Document No. 32-5052125-00, "Conversion of Pilgrim Nuclear Power Station 1996-2001 Meteorological Data for Use With ARCON96" (16 pages)
4. Proposed Changes to the Pilgrim Technical Specifications - Marked-Up Pages (13 pages)
5. Summary of Commitments
6. Response to NRC Request for Additional Information

cc:

Mr. Robert Fretz, Project Manager
Office of Nuclear Reactor Regulation
Mail Stop: 0-8B-1
U.S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. Robert Walker, Director
Massachusetts Department of Public Health
Radiation Control Program
90 Washington Street
Dorchester, MA 02121

U.S. Nuclear Regulatory Commission
Region 1
475 Allendale Road
King of Prussia, PA 19408

Ms. Cristine McCombs
Mass. Emergency Management Agency
400 Worcester Road
Framingham, MA 01702

Senior Resident Inspector
Pilgrim Nuclear Power Station

<u>Attachment 1 to 2.04.115</u>

Subject: Proposed License Amendment for a Limited Scope Application of the Alternate Source Term Guidelines in NUREG-1465 for Re-evaluation of the Fuel Handling Accident Dose Consequences

List of Tables

List of Tables

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90 and 50.67, Entergy hereby proposes to amend Appendix A, Technical Specifications (TS) of the Pilgrim operating license to change the requirements for handling irradiated fuel and performing core alterations. Specifically, the changes would eliminate operability requirements for secondary containment when handling sufficiently decayed irradiated fuel and performing core alterations, and clarify requirements associated with operations with potential to drain the reactor vessel (OPDRVs). Entergy is proposing to revise the requirements associated with equipment whose performance is not credited in the new calculations.

Pilgrim Technical Specifications currently impose restrictions on plant operations when handling irradiated fuel assemblies or performing core alterations. These restrictions require that certain structures, systems or components (SSCs) be operable. These restrictions assure that the radiological consequences of a fuel handling accident do not exceed those estimated in design-basis analyses.

The changes proposed in this application are consistent with TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," (Reference 20). TSTF-51 removes Technical Specification requirements for engineered-safeguard features (ESF) (e.g., primary/secondary containment, standby gas treatment, isolation capability) to be operable after sufficient radioactive decay has occurred to ensure off-site doses remain below the Standard Review Plan limits. TSTF-51 also deletes operability requirements during core alterations for ESF mitigation features.

A "Fuel Handling Accident" (FHA) (or refueling accident) is discussed in Section 14.5.5 of the Pilgrim Updated Final Safety Analysis Report (UFSAR). UFSAR Table 14.5-5 provides radiological consequences of a fuel handling design basis accident. Attachments 2 and 3 (References 4 and 5) provide supporting calculations for the revised FHA analysis. |

Secondary containment, the standby gas treatment system (SGTS), and control room high efficiency air filtration system (CRHEAFS) mitigate the potential effects of a fuel handling accident and are part of the primary success path for a design-basis FHA. Section 10.17 of the Pilgrim UFSAR describes the main control room environmental control system. The main control room environmental control system has a safety-related subsystem that is referred in Technical Specifications as the CRHEAFS. Section 10.9.3.3 of the UFSAR describes the reactor building ventilation system. Section 5.3 of the UFSAR describes SGTS and secondary containment requirements.

The implementation of these changes could reduce the duration and cost of planned outages while maintaining an adequate safety margin. For example, moving large equipment into secondary containment during an outage must be coordinated with Technical Specification requirements for secondary containment operability. This limits how and when the equipment can be moved, which in turn, can result in delays to certain "critical path" activities and extend outage duration.

Another potential benefit involves the performance of maintenance or repair work on redundant "divisionalized" safety systems. This work is usually scheduled to ensure that one division is operable while work is performed on the other division. Unanticipated problems with the operable division could require the suspension of the movement of irradiated fuel or other core alterations, such as control rod drive testing, until the problem is corrected and the system returned to operable status. The proposed change could also facilitate maintenance or repairs on non-redundant portions of CRHEAFS without suspending refueling activities.

Entergy requests NRC review and approval of the proposed changes by March 1, 2005 to support the Pilgrim refueling outage-15 in April 2005.

2.0 PROPOSED CHANGE

The proposed amendment would change the Limiting Conditions for Operation (LCOs) in Pilgrim Technical Specifications (TS) to relax secondary containment operability requirements when handling fuel that is not "recently" irradiated. The proposed change would allow for more efficient performance of outage work while continuing to provide adequate controls against the release of fission product radioactivity to the outside atmosphere during fuel handling activities.

Current Technical Specifications (Table 3/4.2.D, Specifications 3/4.7.B and 3/4.7.C) require secondary containment, together with SGTS and CRHEAFS to be operable:

(1) During movement of irradiated fuel,
(2) During movement of new fuel over the spent fuel pool,
(3) During core alterations, and
(4) During operations with the potential for draining the reactor vessel (OPDRVs).

Changes are proposed to the secondary containment isolation requirements in TS Table 3/4.2.D and SGTS, CRHEAFS, and Secondary Containment operability requirements specified in TS 3/4.7.B.1 and .2, and TS 3/4.7.C for refueling operations based on the revised FHA. The proposed changes will eliminate operability requirements during fuel handling activities that do not involve "recently" irradiated fuel and during core alterations. The systems will still be required to be operable during OPDRVs and during fuel handling activities involving recently irradiated fuel. The proposed changes maintain requirements for SGTS operability during OPDRVs consistent with Pilgrim License Amendments 166 and 170.

Changes to the TS bases define what time period must elapse before fuel is no longer considered "recently" irradiated. For the current cycle-15 (GE 10x10 fuel type), the minimum time period that must elapse following reactor shutdown is 24 hours. The time period is cycle-specific based upon the type of fuel used, fuel performance during the cycle and the Alternate Source Term (AST) methodology in accordance with 10 CFR 50.67. This is included in the TS Bases to reflect the proposed changes to the Specifications.

2.1 Description of Proposed Changes

Entergy is proposing to modify (a) the licensing basis of the Fuel Handling Accident (FHA) as described in the Pilgrim UFSAR and (b) Technical Specifications for the secondary containment isolation requirements in TS Table 3/4.2.D and operability requirements of SGTS, CRHEAFS, and secondary containment specified in TS 3/4.7.B.1 and .2, and TS 3/4.7.C for refueling operations based on the revised FHA.

1. Licensing Basis Change for Fuel Handling Accident

   Entergy is revising the licensing basis of the FHA described in Section 14.5.5 of the Pilgrim UFSAR. The proposed licensing basis change is the re-evaluation of the FHA using Alternate Source Term methodology and dose consequences analysis in accordance with 10 CFR 50.67, Regulatory Guide (R.G.) 1.183, NUREG-1465, and

TSTF-51. Sections 3.0 and 4.0 of this evaluation provide technical analysis in support of the licensing basis change. Entergy plans to revise Sections 14.5.5 and 5.3 of the UFSAR describing the revised FHA and secondary containment operability requirements upon receipt of the approved license amendment and to submit revised UFSAR Sections pursuant to 10 CFR 50.71(e) (Attachment 6).

2. Technical Specification Bases Changes

Based on TSTF-51, Entergy is revising TS Bases B3/4.7.B.1 describing "recently irradiated fuel assemblies" as fuel that has occupied part of a critical reactor core within the previous 24 hours (i.e. irradiated fuel decayed for 24 hours). This change is derived from the above proposed licensing basis change for postulated fuel handling accident (UFSAR Section 14.5.5) using AST in accordance with 10 CFR 50.67 and R.G. 1.183.

The 24-hour decay period is for the current fuel cycle-15 using the GE fuel Type 14 (10x10). Thereafter, each fuel cycle will be evaluated prior to the succeeding refueling outage to derive the applicable decay period for the fuel cycle. If required, the TS Bases will be revised in accordance with 10 CFR 50.59 to ensure Pilgrim remains in compliance with the secondary containment isolation and operability requirements of SGTS and CRHEAFS for movement of recently irradiated fuel assemblies.

3. Technical Specification Changes

Analyses of the radiological dose analysis of a postulated FHA involving irradiated fuel assembles that have been allowed to decay for at least 24 hours show that the calculated total effective dose equivalent (TEDE) values to the control room occupants and at the exclusion area boundary without crediting Secondary Containment, SGTS and CRHEAFS operations are below the allowable TEDE limits established in 10 CFR 50.67 (see Section 4.0 of this evaluation). Therefore, after 24 hours, movement of irradiated fuel assemblies can commence and continue without the operability requirements for SGTS, CRHEAFS, and Secondary Containment. This conclusion (see Section 4.4 of this evaluation) forms the basis for the proposed TS changes. It is not considered credible for fuel movement to commence prior to the 24-hour decay time elapsing.

Since secondary containment operability is not required during the movement of fuel assemblies that are not recently irradiated, the secondary containment isolation requirements (TS 3/4.2.D, Table 3.2.D), and operability requirements for SGTS (TS 3/4.7.B.1), CRHEAFS (TS 3/4.7.B.2), and secondary containment (TS 3/4.7.C) are revised to be applicable only when handling recently irradiated fuel assemblies, consistent with TSTF-51.

The requirement of "movement of new fuel over the spent fuel" is deleted in Specifications 3.7.B.1.a, .c, and .e; 3.7.B.2.a, .c, and .e, and 3.7.C. because the consequences of an accident involving movement of new fuel over the spent fuel are bounded by the analysis performed for irradiated fuel.

The operability requirements during CORE ALTERATIONS for ESF mitigation features are deleted as part of this proposed license amendment. The accidents postulated to occur during core alterations, in addition to the postulated FHA, are inadvertent criticality due to control rod removal error and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. Except for damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in significant radioactive release is the fuel handling accident, the proposed TS changes omitting CORE ALTERATIONS is justified.

License Amendments 166 and 170 revised the operability requirements of secondary containment, SGTS, and CRHEAFS for startup, run, and hot shutdown modes, during movement of irradiated fuel assemblies, during movement of new fuel over the spent fuel pool, during core alterations, and during OPDRVs. The revised FHA does not alter the operability requirements for OPDRV conditions. Thus, operability requirements for OPDRVs are retained. Operability requirements for OPDRVs are added in TS Table 3.2.D Actions A and B, and Specifications 3/4.7.B.1.e (SGTS) and 3/4.7.B.2.e (CRHEAFS) since OPDRVs events can be postulated to cause fission product release to the secondary containment. Since secondary containment is the only barrier to release of fission products to the environment, secondary containment operability is required during OPDRVs. Thus, if one train of SGTS and CRHEAFS is not operable during OPDRVs, actions must be initiated to suspend OPDRVs. Accordingly, requirements for operability during OPDRVs are added to Specifications 3/4.7.B.1.e (SGTS) and 3/4.7.B.2.e (CRHEAFS). These additional requirements are more restrictive and are consistent with the License Amendments 166 and 170.

The proposed Pilgrim TS changes follow TSTF-51 and recently approved James A. Fitzpatrick and Duane Arnold license amendments. Even though Pilgrim's current TS appear different in style than the TSTF-51, the proposed changes are consistent with TSTF-51 requirements, and approved James A. Fitzpatrick and Duane Arnold TS changes in scope and requirements.

A summary of the proposed TS changes are presented in Table 1:

TABLE 1 - Summary of Proposed Changes to the Technical Specifications

Technical Specification Section	Title	Add "recently" irradiated ?	Delete CORE ALTERATIONS?	Operations with a potential for Draining the Reactor Vessel Requirement Clarified?
3/4.2.D.	Secondary Containment Isolation and SGTS Actuation Instrumentation, (Table 3.2.D)	YES	N/A	YES
3/4.7.B.1.	Standby Gas Treatment System (SGTS)	YES	YES	YES
3/4.7.B.2.	Control Room High Efficiency Air Filtration System (CRHEAFS)	YES	YES	YES
3/4.7.C	Secondary Containment	YES	YES	YES

The proposed TS changes are discussed below:

a. TS Table 3.2.D (TS Page 3/4.2-24):

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

Current TS	Proposed TS (See Note below)
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~~ *movement of recently irradiated fuel assemblies and operations with potential to drain the reactor vessel (OPDRVs).* B. Isolate secondary containment and start the standby gas treatment system *during movement of recently irradiated fuel assemblies and OPDRVs.*

Notes: Proposed changes are indicted in italics.

This proposed change does not alter the isolation set points or capability of the secondary containment, but provides consistency for the secondary containment operability requirements in accordance with TSTF-51. OPDRVs is added to the Actions A and B to be consistent with TS 3/4.7.B.1.e.

b. TS 3/4.7.B.1.a, .c, and .e (TS pages 3/4.7-11, 12,and 13) for SGTS:

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

Current TS	Proposed TS
1. Standby Gas Treatment System 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),	1. Standby Gas Treatment System 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 *recently* 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),

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

Current TS	Proposed TS
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. 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.	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. 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*~~.

<u>Notes</u>:

The consequences analysis for the postulated FHA has shown that SGTS is not required for the handling of fuel assemblies that are not recently irradiated. The requirements of 3/4.7.B.1.e (SGTS) have been expanded to include handling of recently irradiated fuel assemblies whether or not the plant is operating at the time.

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

Current TS	Proposed TS
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. If the system is not made fully operable within 7 days, i) place the operable train in operation immediately <u>or</u> ii) suspend movement of irradiated fuel assemblies in secondary containment or new fuel handling over the spent fuel pool or core. Any fuel assembly movement in progress may be completed.	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~~, *movement of recently irradiated fuel assemblies and operations with a potential for draining the reactor vessel (OPDRVs)* 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. If the system is not made fully operable within 7 days, i) place the operable train in operation immediately. <u>OR</u> ii) suspend movement of *recently* irradiated fuel assemblies in secondary containment *and initiate actions to suspend OPDRVs* ~~or new fuel handling over the spent fuel pool or core~~. Any fuel assembly movement in progress may be completed.

<u>Notes:</u>

The insertion of operability requirements for OPDRVs is consistent with License Amendment 166 and 170 and is added to ensure actions are clear. The consequences analysis for the postulated FHA has shown that SGTS is not required for the handling of fuel assemblies that are not recently irradiated.

c. <u>TS 3/4.7.B.2.a (TS pages 3/4.7-14) for CRHEAFS:</u>

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

Current TS	Proposed TS
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),	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 *recently* 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),

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

Current TS	Proposed TS
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.	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.*~~

<u>Notes</u>:

The consequences analysis for the postulated FHA has shown that CRHEAFS is not required for the handling of fuel assemblies that are not recently irradiated. The requirements of 3/4.7.B.2.e (CRHEAFS) have been expanded to include handling of recently irradiated fuel assemblies whether or not the plant is operating at the time.

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

Current TS	Proposed TS
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. If the system is not made fully operable within 7 days, i) perform surveillance 4.7.B.2.b.4 for the operable CRHEAF every 24 hours <u>or</u> ii) suspend movement of irradiated fuel assemblies in secondary containment or new fuel handling over the spent fuel pool or core. Any fuel assembly movement in progress may be completed.	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*~~, *movement of recently irradiated fuel assembles and operations with a potential for draining the reactor vessel (OPDRVs)* 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. If the system is not made fully operable within 7 days, i) perform surveillance 4.7.B.2.b.4 for the operable CRHEAF every 24 hours. <u>OR</u> ii) suspend movement of *recently* irradiated fuel assemblies in secondary containment *and initiate actions to suspend OPDRVs* ~~*or new fuel handling over the spent fuel pool or core*~~. Any fuel assembly movement in progress may be completed.

<u>Notes</u>:

The insertion of operability requirements for OPDRVs is consistent with License Amendment 166 and 170 and is added to ensure actions are clear. The consequences analysis for the postulated FHA has shown that CRHEAFS is not required for the handling of fuel assemblies that are not recently irradiated.

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

Current TS	Proposed TS
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).	1. Secondary containment shall be OPERABLE when in the Run, Startup and Hot Shutdown MODES, during movement of *recently* 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).
2. a. With Secondary Containment inoperable when in the Run, Startup and Hot Shutdown MODES, restore Secondary Containment to OPERABLE status within 4 hours. b. Required Action and Completion Time of 2.a not met, be in HOT Shutdown in 12 hours AND Cold Shutdown within 36 hours. 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 1. Suspend movement of irradiated fuel assemblies in the secondary containment. AND 2. Suspend movement of new fuel over the spent fuel pool. AND 3. Suspend CORE ALTERATIONS AND 4. Initiate action to suspend OPDRVs.	2. a. With Secondary Containment inoperable when in the Run, Startup and Hot Shutdown MODES, restore Secondary Containment to OPERABLE status within 4 hours. b. Required Action and Completion Time of 2.a not met, be in HOT Shutdown in 12 hours AND Cold Shutdown within 36 hours. c. With Secondary Containment inoperable during movement of *recently* irradiated fuel assemblies, and ~~during movement of new fuel over the spent fuel pool, and during CORE ALTERATIONS, and~~ during OPDRVs, immediately: 1. Suspend movement of *recently* irradiated fuel assemblies in the secondary containment. AND ~~2. Suspend movement of new fuel over the spent fuel pool.~~ ~~AND~~ ~~3. Suspend CORE ALTERATIONS~~ ~~AND~~ ~~4~~2. Initiate actions to suspend OPDRVs.

Notes:

The consequences analysis for the FHA has shown that SGTS, CRHEAFS, and secondary containment operability are not required for the handling of fuel assemblies that are not recently irradiated.

e. TS Bases Changes for B3/4.7.B.1 and .2, and B3/4.7.C (pages B3/4.7-10,12, and 13)

The Bases for SGTS are revised to include a description of "recently" irradiated fuel assembles. The Bases for SGTS, CRHEAFS, and Secondary Containment are revised describing the revised operability requirements for movement of fuel assemblies that are not recently irradiated based upon proposed TS changes. These revised pages are attached for information only. Entergy will revise and issue these Bases pages as part of implementation of the approved license amendment.

Attachments 4 and 5 provide marked-up pages and re-typed pages of TS and TS Bases. Attachment 6 provides a summary of commitments.

3.0 BACKGROUND AND BASIS FOR PROPOSED TS CHANGE

The NRC issued a new regulation, 10 CFR 50.67, in December 1999, which provides a means for power reactor licensees to replace their existing accident source term with an alternate source term (AST) (Reference 1). Regulatory Guide 1.183 (Reference 2) provides guidance for the implementation of ASTs. Regulation 10 CFR 50.67 requires licensees seeking to use ASTs to apply for a license amendment and include an evaluation of the consequences of the affected design-basis accidents. This application addresses these requirements by proposing limited scope application of the AST described in R.G. 1.183 in evaluating the radiological consequences of an FHA. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR 50 Appendix A GDC-19.

4.0 TECHNICAL ANALYSIS

4.1 Alternate Source Term

Entergy has completed a calculation following 24 hours decay time evaluating the potential dose consequences of the fuel handling accident. A copy of the calculation is included with the application package (Attachment 2). This calculation uses the AST guidelines outlined in NUREG-1465 (Reference 1), Regulatory Guide 1.183 (Reference 2) and Regulatory Guide 1.194 (Reference 18). This calculation demonstrates that radiological doses at the exclusion area boundary (EAB), low population zone (LPZ) and in the control room (CR) are within allowable limits without crediting secondary containment operability, control room high efficiency air filtration or standby gas treatment systems, after a 24-hour fuel decay period following reactor shutdown.

4.2 Atmospheric Dispersion (X/Q)

Atmospheric dispersion factors (X/Q's) at the normal (primary) control room air intake were calculated using the ARCON96 (Atmospheric Relative CONcentrations in Building Wakes, Reference 17) (see Attachment 3) as follows:

- 5 years' worth of hourly meteorological data collected at the Pilgrim site.
- The ARCON 96 computer code for atmospheric transport of the released radioactivity to the control room fresh air intake (R.G. 1.194).
- The AEOLUS-3 computer code for atmospheric transport of the released

radioactivity to offsite receptors (which implements the guidance of R.G. 1.145, Rev. 1).

Primary assumptions used in the analysis (Attachment 3) for two release locations are summarized as follows:

Reactor Building Vent Release Point

- The reactor building is assumed to be open during the refueling operations, with the normal reactor building ventilation on, such that all releases to the environment would be via the reactor building vent. (The potential leakage via the reactor building siding is very unlikely in view of the structural design of the reactor building walls above the refueling floor.)
- The reactor building vent was evaluated as a ground-level release for off-site receptor locations for EAB and LPZ dose calculations.
- Actual reactor building vent elevation was used as the release point for control room receptor based on R.G. 1.194, as a ground level release.
- The reactor building truck airlock doors were assumed to be closed. These doors are normally closed and are arranged in an "air lock" configuration that allows passage by opening only one door at a time.

Reactor Building Truck Airlock Door Release Point

Reactor building truck lock door release point was evaluated and found to lead to lower control room doses due to the lower atmospheric dispersion factor from the truck lock to the control room fresh air intake.

Control Room Normal Ventilation Intake Flow of Fresh Air

The control room air intake rate was assumed to be 1000 cfm (a low value), and 9000 cfm (a high value bounding the approximate 7200 cfm normal fresh air intake flow to the control room) to show that there is little impact on the control room dose for any flow within the range.

Table 2 summarizes the results of the X/Q calculations.

TABLE 2 - Atmospheric Dispersion Factors (X/Q's)
for Control Room X/Q (s/m^3) (from Table 3.3 of Attachment 2)

Time Interval (hrs.)	Reactor Building Vent Release point	Reactor Building Truck Airlock Door Release Point
0 - 2	1.76E-03	9.72E-04
2 - 8	1.25E-03	7.52E-04
8 - 24	4.26E-04	2.80E-04
24 - 96	3.67E-04	1.93E-04
96 - 720	3.15E-04	1.61E-04

As noted in the introduction of R.G. 1.194, many of the positions in the guide represent significant changes. ARCON96 implements an improved building wake dispersion algorithm; assessments of ground level, building vent, elevated and diffuse-source release models; use of hour-by-hour meteorological observations; sector averaging; and directional dependence of dispersion conditions. Therefore, no discussion of the comparison with current licensing basis X/Q values is presented.

4.3 Radiological Consequences of a Design-Basis Fuel Handling Accident

The radiological consequences of a design-basis FHA were analyzed using Pilgrim-specific design inputs and assumptions. Evaluations were performed without the ESF functions after 24 hours of reactor fuel decay. The calculations assumed that the main control room ventilation remained in its normal (non-emergency) mode with no SGTS or CRHEAFS in operation. Plant-specific design inputs were validated (See NEI 99-03, Reference 8) to ensure that they are representative of "as-built" plant design conditions.

Primary assumptions used in this analysis are summarized below:

- Alternate source terms used
- No credit taken for ESF systems (secondary containment operability, standby gas treatment system filtration or operation of the control room high efficiency air filtration system) in the applicable calculations
- All releases are unfiltered through the reactor building vent
- Analyses used guidance in Appendix B of Regulatory Guide 1.183
- Fuel decayed for a period of 24 hours
- Release occurred during a 2-hour period
- Credited scrubbing of the halogen activity by water over dropped assembly

Table 3 summarizes the key assumptions and design-basis parameters used in the development of the source term. The TEDE doses for the exclusion area boundary (EAB), low population zone (LPZ) and control room (CR) were calculated using the post-FHA release through the reactor-building vent for 0-2 hours using the newly calculated X/Qs (Attachment 3). It should be noted that FHA 0-2 hours release commences after a fuel decay period of 24 hours following shutdown.

Appendix B of RG 1.183

Appendix B of Regulatory Guide 1.183 (Reference 2) outlines six groups of assumptions acceptable to the NRC staff for evaluating the radiological consequences of a design-basis fuel handling accident. The following sections will discuss these assumptions as they relate to the new analyses.

- Source Term

The fractions of core inventory assumed to be in the gap for the various nuclides are taken from Table 3 "Non-LOCA Fraction of Fission Product Inventory in Gap" of Regulatory Guide 1.183. These release fractions were then applied to the core fission product inventory, a conservative estimate of 151 damaged fuel rods, and a maximum core radial peaking factor of 2.1, to produce the source term used in the analysis.

- Water Depth

A decontamination factor (DF) of 200 was assumed for the scrubbing effects of water on halogen activity released. The DF was based on a minimum of 23 feet of water over the dropped assembly. While the minimum water depth above spent fuel assemblies in the spent fuel pool permitted by TS is less (18 ft), calculations show that as a result of a reduced drop height, an assembly dropped over the spent fuel pool would involve less energy and result in fewer damaged fuel rods. The normal depth of water in the spent fuel pool provides about 23 feet of coverage. Consequently, the radiological consequences of a FHA over the reactor vessel bound the consequences of a FHA over the spent fuel pool even with the reduced scrubbing.

- Noble Gases

A DF of 1 was used because the retention of noble gases in the water in the fuel pool or reactor cavity is negligible. Particulate radionuclides were assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

- Fuel Handling Accidents within the Fuel Building

This section of the regulatory guide is not applicable, as Pilgrim does not have a separate fuel building.

- Fuel Handling Accident within Containment

Entergy analyses assumed that the normal reactor building refuel floor ventilation system is functioning and the exhaust dampers are open during fuel handling operations. No credit has been taken for ESF actuation or manual actions to restore secondary containment closure. Radioactive material that escapes from the spent fuel pool, or reactor cavity, is released to the environment over a 2-hour period, following a fuel decay period of 24 hours following reactor shutdown.

Credit for dilution or mixing of the activity released from the reactor cavity or spent fuel pool by natural or forced convection inside the containment was not considered.

- Core Inventory

The core inventory is based on a thermal power level of 2028 MWt, plus a measurement uncertainty of 0.5%. A radial peaking factor of 2.1 was used instead of 1.5 as recommended in Regulatory Guide 1.183 to provide additional margin for future core reloads and different fuels. The isotopic activities released from the damaged fuel rods are calculated based on the number of rods failed during the FHA and core thermal power level.

- Number of Fuel Rods Damaged

The analyses assumed that 151 fuel rods were damaged. The type of fuel used was GE Type 14 (10x10). Refer to GNF report NEDE-31152P (Reference 15) and NEDE-24011-P-A-US-14 (Reference 16) for additional information.

- Timing of Release Phase

Gap activity in the damaged rods was assumed to be released instantaneously. The analysis assumed that the release to the atmosphere would occur over a 2-hour period, following a 24-hour fuel decay following reactor shutdown.

RADTRAD Computer Code

The radiological evaluation of postulated FHA was carried out through the use of the ELISA-2 computer code (see Attachment C of Attachment 2). An alternative analysis was performed using the RADTRAD (RADionuclide Transport and Removal and Dose Estimation) computer program, Version 3.02 (Reference 12) for comparison with ELISA-2 results (see Attachment A of Attachment 2). The vendor does not have Q-version of RADTRAD in their computer software index for use in safety-related applications, as such the RADTRAD results are provided for information only.

Control Room Envelope In-leakage

Infiltration pathways, other than through the normal CR outside air intake, were not considered in this analysis because the control room ventilation system was assumed to operate in it's "normal" (non-emergency Mode) without taking credit for emergency filtration systems (i.e. CRHEAFS) or the effects of pressurizing or isolating the control room envelope.

4.4 Results of Fuel Handling Accident Dose Consequences

The resulting doses at the EAB, LPZ, and CR locations are compared with the regulatory allowable limits in Table 4. Table 5 compares these to current licensing basis (CLB) radiological doses for a refueling accident (or FHA). The CLB doses are from Table 14.5-5 of the Pilgrim UFSAR.

The dose evaluations of the postulated fuel handling accident demonstrates that the calculated TEDE values to the control room, EAB, and LPZ using 24-hour decay time for the reactor fuel with no operable SGTS and CRHEAFS and with normal unfiltered control room and with the reactor building ventilation system in operation or with the reactor building truck lock doors open are less than the acceptance values specified in 10 CFR 50.67.

The FHA dose consequence analysis considered the reactor building exhaust vent as the release point with normally closed truck airlock doors with no credit for the operability of secondary containment, SGTS, or CRHEAFS. Pilgrim also considered release through the open reactor building truck airlock doors with no credit for the operability of secondary containment, SGTS, or CRHEAFS. The resulting dose consequences are also below the regulatory limits and are bounded by the dose consequences with the reactor building exhaust vent as the release point with normally closed airlock doors. Thus, the reactor building truck airlock doors can be open during the movement of non-recently irradiated fuel assemblies without adversely impacting the FHA dose consequence analysis.

4.5 Core Alterations

The accidents postulated to occur during core alterations, in addition to the fuel handling accident, are inadvertent criticality due to 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. Therefore, the only accident postulated to occur during core alterations that result in significant radioactive release is the fuel handling accident. Thus, the consequence of a fuel handling accident envelops the consequences of potential accidents postulated to occur during core alterations. Therefore, the proposed TS changes omitting core alterations are justified.

4.6 Operations with a Potential for Draining the Reactor Vessel (OPDRVs)

Secondary containment operability is required during OPDRVs, since OPDRV events can be postulated to cause fission product release, different than the fuel handling accident. The revised FHA does not alter the operability requirements for OPDRV conditions. Since secondary containment is the only barrier to the release of fission products into the environment, the SGTS is required to be operable to maintain the secondary containment operability during OPDRVs and if one train of SGTS is not operable during OPDRVs, actions must be initiated. Similarly, CRHEAFS is required to assure control room habitability during OPDRVs and if one train of CRHEAFS is not operable, actions must be initiated. License Amendments 166 and 170 revised the operability requirements of secondary containment, SGTS, and CRHEAFS for startup, run, and hot shutdown modes, during movement of irradiated fuel assemblies and during OPDRVs. Thus, TS 3/4.7.B.1.e (SGTS) and 3/4.7.B.2.e (CRHEAFS) are revised such that if one train of SGTS and CRHEAFS are not operable during OPDRVs, actions must be initiated to suspend OPDRV events. These revised requirements are more restrictive than the wording of the current Technical Specifications, but are consistent with the License Amendments 166 and 170.

TABLE 3 - Key Inputs for Fuel Handling Analysis
(Extracted from Attachment 2)

Input/Assumption	Value
Fuel Type	General Electric GE14
Reactor Thermal Power	2038 MWt
Initial Mass of Uranium	101.8-105.56 metric tons of uranium
Initial Core Average Enrichment	3.9 – 4.6 w%[1] U-235
End of Cycle Core Wide Exposure	5 – 60 GWD/MT[1]
Number of assemblies in the core	580
Total number of fuel rods in GE14 fuel (10x10) assembly	92
Assembly Average Operating Power Level (2038MWt/580 assemblies)	3.514 MWt/assembly
Number of Failed Rods	151
Radial Peaking Factor	2.1
Equivalent Number of Damaged Peak Assembles and Fraction there of (151 rods/92rods per assembly	1.641
Power Level Associated with damaged Rods in Peak Assembly (3.514 MWtx1.641 assemblies x 2.1 Peaking Factor)	12.11 MWt
Fuel Decay Period	24 hours
Release Fractions	RG 1.183 Table 3
Effective pool iodine decontamination factor	200
Iodine species fractions in airborne release (composition above pool)	Elemental = 57% Organic = 43%
Noble Gas decontamination factor	1
Release duration (starting after 24-hour fuel decay	2 hrs
Release location: No secondary containment (No SGTS)	Reactor Building Vent

[1] Range considered in determining bounding source term

TABLE 4 - Radiological Dose Effects of Fuel Handling Accident (Attachment 2)

	Receptor Location		
	Control Room Dose in 30 days with normal unfiltered control room ventilation system in operation and with no CRHEAFS	EAB In 2 hours	LPZ In 30 days
Calculated TEDE Dose (rem) with 24-hr decay AST source, w/o SGTS	2.846, with 1000cfm air intake 2.863, with 9000cfm air intake	1.439	0.92
Allowable TEDE Limit (rem)	5.00	6.30	6.30

TABLE 5 - Comparison of Current Licensing Basis (CLB) and Alternate Source Term (AST) Radiological Doses as a result of a Fuel Handling Accident

Site Boundary / Exclusion Area Boundary		Low Population Zone		Control Room	
CLB (Rem)	AST[2] (Rem)	CLB (Rem)	AST[2] (Rem)	CLB (Rem)	AST[3] (Rem)
24 Thyroid[1]	1.439 TEDE	2.4 Thyroid[1]	0.09 TEDE	X[4]	2.863 TEDE

Notes
1. From Pilgrim UFSAR Table 14.5-5
2. 24-hr decay with no SGTS normal control room and reactor building ventilation, release through the reactor building vent
3. 24-hr decay with no SGTS and no CRHEAFS, normal control room and reactor building ventilation, release through the reactor building vent
4. FHA dose impact on control room not reported in UFSAR

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (Entergy) proposes changes to the (a) Pilgrim licensing basis of the Fuel Handling Accident (FHA) as described in the Pilgrim updated Final Safety Analysis Report (UFSAR) and (b) Pilgrim Technical Specifications (TS). The proposed changes to the TS would eliminate secondary containment operability requirements when handling sufficiently decayed irradiated fuel and performing core alterations, and clarify requirements associated with operations with potential to drain the reactor vessel (OPDRVs). These changes revise secondary containment isolation requirements, and standby gas treatment system (SGTS), control room high efficiency air filtration system (CRHEAFS), and secondary containment operability requirements for refueling operations.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does not involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The proposed changes do not modify the design or operation of equipment used to move spent fuel or to perform core alterations. The proposed changes cannot increase the probability of any previously analyzed accident because they are based on changes in Source Term, atmospheric dispersion and dose consequence analysis methodology, not in procedures or equipment used for fuel handling.

The conservative re-analysis of the fuel handling accident (FHA) concludes that the radiological consequences are within the regulatory limits established 10 CFR 50.67. This conclusion is based on the Alternate Source Term and guidance provided in Appendix B of Regulatory Guide 1.183 and analyses of fission product release and transport path that does not take credit for dose mitigation provided by engineered safeguards including secondary containment, standby gas treatment system (SGTS), and control room high efficiency air filtration system (CRHEAFS). The results of the core alteration events, other than the fuel handling accident, remain unchanged from the original design-basis that showed these events do not result in fuel cladding damage or radioactive release. Since secondary containment is the only barrier to the release of fission products into the environment, the SGTS is required to be operable to maintain the secondary containment operability during OPDRVs and if one train of SGTS is not operable during OPDRVs, actions must be initiated. Similarly, CRHEAFS is required to assure control room habitability during OPDRVs and if one train of CRHEAFS is not operable, actions must be initiated. License Amendments 166 and 170 revised the operability requirements of secondary containment, SGTS, and CRHEAFS for startup, run, and hot shutdown modes, during movement of irradiated fuel assemblies and during OPDRVs. TS 3/4.7.B.1.e (SGTS) and 3/4.7.B.2.e (CRHEAFS) are revised such that if one train of SGTS and CRHEAFS are not operable during OPDRVs, actions must be initiated to suspend OPDRV events. These revised requirements are more restrictive than the wording of the current Technical Specifications, but are consistent with the License Amendments 166 and 170.

Therefore, the proposed changes do not involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed.

2. Does not create the possibility of a new or different kind of accident from any accident previously analyzed?

Response: No

The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

3. Does not involve a significant reduction in the margin of safety?

Response: No

Regulation 10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design-basis radiological consequence analyses. This license amendment application evaluates the consequences of a design-basis fuel handling accident in accordance with this regulation and R.G. 1.183. The revised analysis concludes that the radiological consequences of the fuel handling accident are less than the regulatory allowable limits. Safety margins and analytical conservatisms are retained to ensure the analysis adequately bounds all postulated event scenarios. The selected assumptions and release models provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensates for large uncertainties in facility parameters, accident progression, radioactive material transport and atmospheric dispersion. The proposed TS applicability statements continue to ensure that the TEDE at the boundaries of the control room, the exclusion area, and low population zone boundaries are below the corresponding regulatory allowable limits in 10 CFR 50.67(b)(2). The proposed operability requirements for SGTS and CRHEAFS during OPDRVs are consistent with the License Amendments 166 and 170 and therefore do not reduce the margin of safety.

Therefore, the changes do not involve a significant reduction in margin of safety.

Based on the above, Entergy concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

This section describes how the proposed changes and Entergy's technical analyses satisfy applicable regulatory requirements and acceptance criteria.

10 CFR 50 Appendix A General Design Criterion 61, "Fuel Storage and Handling and Radiological Control"

The general design criteria (GDC) in place today became effective after the Pilgrim construction permit was issued. A September 18, 1992 memorandum to the NRC Executive Director of Operations from the Secretary of the NRC summarized the results of a Commissioners vote in which the Commissioners instructed the NRC staff not to apply the GDC to plants with construction permits issued prior to May 21, 1971. Pilgrim's construction permit was issued on August 26, 1968.

Pilgrim's design and licensing basis for fuel storage and handling and radiological controls is detailed in the updated Final Safety Analysis Report (UFSAR) and other plant-specific licensing basis documents. UFSAR Appendix F provides a comparison of Pilgrim Station with the proposed GDC published by the AEC for public comment in the Federal Register dated July 11, 1967.

<u>10 CFR 50.67 "Accident Source Term"</u>

Regulation 10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design-basis radiological consequence analyses. This document is part of a 10 CFR 50.90 license amendment application and evaluates the consequences of a design-basis fuel handling accident previously reported in the safety analysis report.

<u>10 CFR 50.65 "Requirements for monitoring the effectiveness of maintenance at nuclear power plants"</u>

Regulation 10 CFR 50.65(a)(4) requires licensees to assess and manage changes in risk that result from taking risk-significant systems out-of-service or during certain maintenance activities. The NRC staff, in Regulatory Guide 1.182 (Reference 19), states that the methods detailed in Section 11 of NUMARC 93-01 (Reference 13) are acceptable for complying with the requirements of 10 CFR 50.65(a)(4). Section 11.3.6.5 "Containment - Primary (PWR)/Secondary (BWR)," of NUMARC 93-01 states:

> Maintenance activities involving the need for open containment should include evaluation of the capability to achieve containment closure in sufficient time to mitigate potential fission product release. This time is dependent on a number of factors including the decay heat level and the amount of RCS inventory available.
>
> For BWRs, Technical Specifications may require secondary containment to be closed under certain conditions such as during fuel handling and operations with a potential to drain the vessel.
>
> In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, 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 TS operability is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce dose 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.

To further limit the potential radiological consequences of a fuel handling accident at Pilgrim, Entergy will revise the Pilgrim guidelines for assessing systems removed from service during the handling of recently irradiated fuel assemblies or core alterations to implement the provisions of Section 11.3.6.5 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3 (Reference 13). These guidelines will address the capabilities to promptly close secondary containment and will be completed prior to the implementation of this license amendment. (This commitment is also consistent with the NRC-approved generic TS change, TSTF-51 (Reference 20) regarding usage of the term "recently irradiated fuel assemblies.")

<u>10 CFR 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance"</u>

This paragraph provides criteria for evaluating the radiological aspects of reactor sites. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. A similar footnote appears in 10 CFR 50.67.

In accordance with the provisions of 10 CFR 50.67(a), the radiation dose reference values in 10 CFR 50.67(b)(2) were used in these analyses in lieu of those prescribed in 10 CFR 100. (Refer to footnote 5 on page 1.183-7 of Regulatory Guide 1.183, dated July 2000.)

<u>Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", March 1972</u>

Regulatory Guide 1.25 is not applicable to the application. Regulatory Guide 1.183 supersedes corresponding radiological assumptions provided in other regulatory guides and standard review plan chapters when used in conjunction with an approved alternate source term and the TEDE criteria provided in 10 CFR 50.67.

<u>Regulatory Guide 1.183, "Alternative Radiological Source Terms for evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000</u>

Regulatory Guide 1.183 outlines acceptable applications of ASTs, the scope, nature and documentation of associated analyses and evaluations, consideration of impacts on analyzed risk; and content of submittals. It also establishes acceptable ASTs and identifies the attributes of ASTs acceptable to the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the AST.

Entergy used this regulatory guide extensively in the preparation of this "limited scope implementation" evaluation, the supported application and the supporting analyses. This application and the supporting analyses comply with this guidance to the extent practical.

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"

NUREG-1465 (Reference 1) provides more realistic estimates than TID-14844 (Reference 14) of "source term" releases into containment in terms of timing, nuclide types, quantities, and chemical form, given a severe core melt. NUREG-1465 provides much of the technical basis for the regulatory positions in Regulatory Guide 1.183.

NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents"

This SRP section covers the review of the radiological effects of a postulated fuel handling accident. Revision 1 does not reflect the guidance in Regulatory Guide 1.183 or the promulgation of 10 CFR 50.67.

5.3 Conclusion

The results of these analyses indicate that the dose at the exclusion area boundary (EAB) would be no more than 4.9 rem total effective dose equivalent (TEDE) and the dose at the low-population zone (LPZ) would be no more than 0.05 rem TEDE. These results are less than the TEDE criteria of 6.3 rem set forth in Regulatory Guide 1.183 and are a small fraction of the dose criteria in 10 CFR 50.67(b)(2)(i) and (ii). The analyses also show that control room operators would receive no more than 2.9 rem TEDE. These doses are less than the TEDE limit of 5 rem contained in 10 CFR 50.67(b)(2)(iii) and GDC-19, "Control Room."

In conclusion, based on the considerations discussed above,

(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.

The NRC has approved similar TS changes (References 3 and 22).

6.0 ENVIRONMENTAL CONSIDERATION

Entergy has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve

(i) a significant hazards consideration,
(ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or
(iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," L. Soffer et al., February 1995.

2. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors," July 2000.

3. NRC letter, D. S. Hood to G. Van Middlesworth, dated April 16, 2001 regarding "Duane Arnold Energy Center - Issuance of Amendment Regarding Secondary Containment Operability During Movement of Irradiated Fuel and Core Alterations (TAC No. MB1569)."

4. Areva Document No. 32-5052589-01, "Radiological Consequences of a Design-Basis Fuel Handling Accident Based on the Alternate Source Term Methodology", (141 pages)

5. Areva Document No. 32-5052821-01, "Determination of Atmospheric Dispersion Factors for Accident Analyses Using Reg Guide 1.145 and 1.194 Methodologies" (80 pages); Areva Document No. 32-5052036-00, "Evaluation of Pilgrim Nuclear Power Station 1996-2001 Meteorological Data" (32 pages); and Areva Document No. 32-5052125-00, "Conversion of Pilgrim Nuclear Power Station 1996-2001 Meteorological Data for Use With ARCON96" (16 pages)

6. 10 CFR 50.67, "Accident Source Term"

7. J. N. Hamawi, "ELISA-2 – A Software Package for the Radiological Evaluation of Licensing and Severe Accidents at Light-Water Nuclear Power Plants Based on the Classical and Alternative-Source-Term Methodologies," ENTECH Engineering, Inc., Techncial Document P100-R22 (Vols A-G, Version 2.2, May 2002

8. NEI 99-03, "Control Room Habitability Guidance."

9. GE Technical Report NEDO-20360, "Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Application for 8x8 Fuel", Rev. 1, November 1974.

10. Global Nuclear Fuel, NEDE-31152P, Revision 8, Class III, April 2001, "General Electric Fuel Bundle Designs."

11. Regulatory Guide 1.25, "Assumptions Used For Evaluating The Potential Radiological Consequences Of A Fuel Handling Accident In The Fuel Handling and Storage Facility For Boiling and Pressurized Water Reactors," dated March 23, 1972.

12. S. L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, USNRC, April 1998.

13. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.

14. J. J. DiNunno et. al., "Calculation of Distance Factors for Power and Test Reactor Sites," USAEC TID-14844, U. S. Atomic Energy Commission (now USNRC), 1962.

15. Global Nuclear Fuel, NEDE-31152P, Revision 8, Class III, April 2001, "General

Electric Fuel Bundle Designs."

16. General Electric, NEDE-24011-P-A--US-14, "General Electric Standard Application for Reactor Fuel (Supplement for United States)."

17. ARCON96 computer code described in NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997.

18. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants", June 2003.

19. Regulatory Guide 1.182, "Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants."

20. TSTF-51, Rev. 2 , "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Excel Services Corporation.

21. 10 CFR 100.11 "Determination of exclusion area, low population zone and population center distance."

22. NRC letter, G. S. Vissing to M. Kansler, dated September 12, 2002 regarding "James A FitzPatrick Nuclear Power Plant – Amendment Re: Technical Specification Change to the Requirements for Handling Irradiated Fuel Assemblies (TAC No. MB5328)."