May 1, 2006

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE, LLC and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-OLA
	)	ASLBP No. 04-832-02-OLA
(Vermont Yankee Nuclear Power Station)	)	

NRC STAFF'S ANSWER TO NEW ENGLAND COALITION'S
REQUEST FOR LEAVE TO FILE NEW CONTENTIONS

ATTACHMENT 5

OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW STANDARD FOR EXTENDED POWER UPRATES

APPROVED BY: /RA/
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RS-001, Revision 0
DECEMBER 2003

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PURPOSE

The purpose of this review standard is to provide guidance for the Nuclear Regulatory Commission (NRC) staff's review of extended power uprate (EPU) applications to enhance consistency, quality, and completeness of reviews.

This review standard also informs licensees of the guidance documents the staff uses when reviewing EPU applications. These documents provide acceptance criteria for the areas of review. This should allow licensees to prepare EPU applications that are complete with respect to the areas that are within the staff's scope of review. To further improve the efficiency of the staff's review of EPU applications, licensees are encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants.

Use of this review standard should not undermine the NRC's longstanding topical report review and approval process. If a licensee references an NRC-approved topical report for an area covered by this review standard, the staff will review the application only to ensure that the licensee is applying the topical report under conditions for which the topical report was approved, using appropriate plant-specific inputs.

The staff will review plants against their design bases. Licensees are encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants. This should help the staff identify areas where the criteria and/or guidance in the review standard does not apply to the plant under review. The staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard.

In addition to this review standard, the NRC maintains a Web site on power uprates at http://www.nrc.gov/reactors/operating/licensing/power-uprates.html. Some of the material on this Web site includes:

- the status of completed, ongoing, and expected power uprate reviews
- general guidance related to power uprates
- references to publicly available correspondence related to reviews of recently completed power uprates (including licensees' responses to NRC staff requests for additional information, as well as NRC staff safety evaluations)

BACKGROUND

Facility operating licenses and technical specifications specify the maximum power level at which commercial nuclear power plants may be operated. NRC approval is required for any changes to facility operating licenses or technical specifications. The process for making changes to facility operating licenses and technical specifications is governed by Title 10 of the *Code of Federal Regulations*, Part 50.

The process of increasing the licensed power level at a commercial nuclear power plant is called a "power uprate." Power uprates are categorized based on the magnitude of the power increase and the methods used to achieve the increase. Measurement uncertainty recapture power uprates result in power level increases that are less than 2 percent and are achieved by implementing enhanced techniques for calculating reactor power. Stretch power uprates typically result in power level increases that are up to 7 percent and do not generally involve major plant modifications. EPUs result in power level increases that are greater than stretch power uprates and usually require significant modifications to major plant equipment. The NRC has approved EPUs for increases as high as 20 percent. This review standard is applicable to EPUs.

This review standard establishes standardized review guidance and acceptance criteria for the staff's reviews of EPU applications to enhance the consistency, quality, and completeness of reviews. It serves as a tool for the staff's use when processing EPU applications in that it provides detailed references to various NRC documents containing information related to the specific areas of review.

This review standard also informs licensees of the guidance documents the staff will use when reviewing EPU applications. This will help licensees prepare EPU applications that address those topics necessary for a complete application. By addressing the areas in the review standard, a licensee could prepare and submit a more complete application and thus minimize the staff's need for requests for additional information (RAIs). This would improve the efficiency of the staff's reviews.

The development of this review standard included an evaluation of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), to determine the applicability and adequacy of the various SRP sections to the review of EPU applications and development/revision of guidance, as necessary. During this evaluation, the staff considered the versions of the SRP sections identified in the matrices in Section 2 of this review standard. To determine the need for guidance beyond that in the SRP, the staff reviewed: (1) safety evaluations for previously approved power uprates, (2) previously approved topical reports for EPUs, (3) various reports related to lessons learned from the Maine Yankee experience (e.g., Report of the Maine Yankee Lessons Learned Task Group, dated December 1996), and (4) generic communications. The staff also considered feedback from internal and external stakeholders. In addition, the staff reviewed RAIs issued for recent EPU applications to ensure that the review standard adequately addresses areas where repeat RAIs have been issued.

The staff reviewed NRC procedural guidance documents to identify those applicable to processing EPU applications. The review of these documents also included consideration of the recommendations in various reports related to the Maine Yankee experience and the feedback received from internal and external stakeholders.

Figure 1 provides a graphical representation of the development of the review standard.

GUIDANCE

This review standard provides guidance for

- processing EPU applications (Section 1)
- performing technical reviews (Section 2)
- preparing safety evaluations to document the reviews (Section 3)

This review standard also includes a reference to the NRC's Inspection Manual, which provides guidance for conducting inspections related to the implementation of power uprates (Section 4).

SECTION 3.2 of RS-001

TEMPLATE SAFETY EVALUATION

for

BOILING-WATER REACTOR EXTENDED POWER UPRATE

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. ___ TO FACILITY OPERATING LICENSE NO. **[XXX-XX]**

[NAME OF LICENSEE]

[NAME OF FACILITY]

DOCKET NO. 50-**[XXX]**

1.0 INTRODUCTION

1.1 Application

By application dated **[ ]**, as supplemented by letter**[s]** dated **[ ]**, the **[Name of Licensee]** (the licensee) requested changes to the Facility Operating License and Technical Specifications (TSs) for the **[Plant Name]**. The supplemental letter[s] dated [ ], provided additional clarifying information that did not expand the scope of the initial application and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on **[date] (XX FR XXXX)**.

The proposed changes would increase the maximum steady-state reactor core power level from **[current licensed power level]** megawatts thermal (MWt) to **[power level proposed by the licensee]** MWt, which is an increase of approximately **[##]** percent. The proposed increase in power level is considered an extended power uprate (EPU).

1.2 Background

[Plant Name] is a boiling-water reactor (BWR) plant of the BWR/**[#]** design with a Mark-**[#]** containment. **[Plant Name]** has the following special features/unique designs:

[Insert any special features/unique designs]

The NRC originally licensed **[Plant Name]** on **[date]** for operation at **[original licensed power level]** MWt. **[By Amendment No. [###] dated [ ], the NRC granted a power uprate to [Plant Name] of [##] percent, allowing the plant to be operated at [current licensed power level] MWt.] Therefore, the proposed EPU would result in an increase of approximately [##] percent over the original licensed power level [and [##] percent over the current licensed power level] for [Plant Name].]**

1.3 Licensee's Approach

The licensee's application for the proposed EPU follows the guidance in the Office of Nuclear Reactor Regulation's (NRR's) Review Standard (RS)-001, "Review Standard for Extended Power Uprates," to the extent that the review standard is consistent with the design basis of the plant. Where differences exist between the plant-specific design basis and RS-001, the licensee described the differences and provided evaluations consistent with the design basis of the plant. The licensee also used **[Identify topical reports or other documents used by the licensee for guidance related to the scope of the proposed EPU; NRC staff approvals, ranges of applicability, any limitations/restrictions associated with the documents; and consistency of the licensee's application with the ranges of applicability and limitations/restrictions. The discussion in this section is to cover topical reports and other documents referenced for the overall power uprate process. It is not intended to cover topical reports and other documents for specific methods of analyses. Topical reports and other documents referenced for specific methods of analyses are to be covered in the applicable technical evaluation section of this safety evaluation].**

Insert this sentence if the licensee is planning to implement the EPU in one stage.
[The licensee plans to implement the EPU in one step. The licensee plans to make the modifications necessary to implement the EPU during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt starting in Cycle [##].]

Insert this paragraph if the licensee is planning to implement the EPU in stages:
[The licensee plans to implement the EPU in [#] steps of [## and ##] percent. The licensee plans to make modifications necessary to implement the first step during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt during Cycle [##]. The remainder of the modifications will be completed during the refueling outage in [season year (e.g., fall 2003)], with subsequent operation at [##] MWt starting in Cycle [##].]

1.4 Plant Modifications

The licensee has determined that several plant modifications are necessary to implement the proposed EPU. The following is a list of these modifications and the licensee's proposed schedule for completing them.

[Provide a list of plant modifications.]

The NRC staff's evaluation of the licensee's proposed plant modifications is provided in Section 2.0 of this safety evaluation.

1.5 Method of NRC Staff Review

The NRC staff reviewed the licensee's application to ensure 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) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review is to evaluate the licensee's assessment of the impact of the proposed EPU on design-basis analyses. The NRC staff evaluated the licensee's application and supplements. The NRC staff also evaluated **[Include additional review items, as necessary (e.g., audits of certain information at the plant and vendor sites, and independent analyses), for areas where such analyses were deemed appropriate by the NRC staff]**.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered the affects of the changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed EPU conditions. Details of the NRC staff's review are provided in Section 2.0 of this safety evaluation.

Audits of analyses supporting the EPU were conducted in relation to the following topics:

[Provide a list of areas for which audits were performed.]

The results of the audits are discussed in section 2.0 of this safety evaluation.

Independent NRC staff calculations were performed in relation to the following topics:

[Provide a list of areas for which independent NRC staff calculations were performed.]

The results of the calculations are discussed in section 2.0 of this safety evaluation.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

SEE INSERT 1 FOR SECTION 3.2 OF RS-001

2.2 Mechanical and Civil Engineering

SEE INSERT 2 FOR SECTION 3.2 OF RS-001

2.3 Electrical Engineering

SEE INSERT 3 FOR SECTION 3.2 OF RS-001

2.4 Instrumentation and Controls

SEE INSERT 4 FOR SECTION 3.2 OF RS-001

2.5 Plant Systems

SEE INSERT 5 FOR SECTION 3.2 OF RS-001

2.6 Containment Review Considerations

SEE INSERT 6 FOR SECTION 3.2 OF RS-001

2.7 Habitability, Filtration, and Ventilation

SEE INSERT 7 FOR SECTION 3.2 OF RS-001

2.8 Reactor Systems

SEE INSERT 8 FOR SECTION 3.2 OF RS-001

2.9 Source Terms and Radiological Consequences Analyses

SEE INSERT 9 FOR SECTION 3.2 OF RS-001

2.10 Health Physics

SEE INSERT 10 FOR SECTION 3.2 OF RS-001

2.11 Human Performance

SEE INSERT 11 FOR SECTION 3.2 OF RS-001

2.12 Power Ascension and Testing Plan

SEE INSERT 12 FOR SECTION 3.2 OF RS-001

2.13 Risk Evaluation

SEE INSERT 13 FOR SECTION 3.2 OF RS-001

3.0 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

To achieve the EPU, the licensee proposed the following changes to the Facility Operating License and TSs for **[Plant Name]**.

[Provide a list of license and TSs changes (including license conditions) and an NRC staff evaluation of each.]

4.0 <u>REGULATORY COMMITMENTS</u>

Insert the following sentence if the licensee has not made any regulatory commitments in support of the EPU.
The licensee has made no regulatory commitments in its application for the EPU.

Insert the following if the licensee has made regulatory commitments in support of the EPU.
The licensee has made the following regulatory commitment(s):

[Provide a summary of each regulatory commitment made by the licensee.]

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 <u>RECOMMENDED AREAS FOR INSPECTION</u>

As described above, the NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. The NRC staff's review has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed EPU. These areas are recommended based on past experience with EPUs, the extent and unique nature of modifications necessary to implement the proposed EPU, and new conditions of operation necessary for the proposed EPU. They do not constitute inspection requirements, but are intended to give inspectors insight into important bases for approving the EPU.

[Provide list of recommended areas for inspection.]

6.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the **[Name of State]** State official was notified of the proposed issuance of the amendment. The State official had **[no]** comments. **[If comments were received, address them here.]**

7.0 <u>ENVIRONMENTAL CONSIDERATION</u>

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register* on **[Date]** (**FR**). The draft Environmental Assessment provided a 30-day opportunity for public comment. *If no comments were received, use the following sentence:* **[No comments were received on the draft Environmental Assessment.]** *If comments were received, use the following sentence*: **[The NRC staff received comments which were addressed in the final environmental assessment.]** The final Environmental Assessment was published in the *Federal Register* on **[Date]** (**FR**). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003.

2. **[Insert additional references as necessary]**

Attachment: List of Acronyms

Principal Contributors:

Date:

INSERT 9

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's **[Updated Safety Analysis Report *or* Updated Final Safety Analysis Report**] related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms.

NOTE: Use Sections 2.9.2 and 2.9.3 below if the licensee's radiological consequences analyses are based on an alternative source term.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

NOTE: There are two cases that may be encountered here: (1) a licensee may be implementing an alternative source term for the first time, or (2) a licensee may have already fully implemented an alternative source term and is revising the previously approved dose analyses that use alternative source term methodologies. The second paragraph for each heading is only needed for a first-time implementation of an alternative source term (either partial or full implementations). Several accidents may have been analyzed - see corresponding SRP sections for further regulatory evaluation text (to be modified), as needed.

Regulatory Evaluation

The NRC staff reviewed the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel handling accident (FHA), control rod drop accident (CRDA), and main steamline break (MSLB). The NRC staff's review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident. Specific review criteria are contained in SRP Section 15.0.1.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff reviewed the implementation of alternative source terms. The NRC's acceptance criteria for implementation of alternative source terms are based on (1) 10 CFR 50.67, insofar as it sets standards for the implementation of an alternative source term in current operating nuclear power plants; (2) 10 CFR 50.49, insofar as it requires qualification of safety-related equipment, as defined in that section, including and based on integrated radiation dose during normal and accident conditions; (3) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident; (4) Paragraph IV.E.8 of 10 CFR Part 50, Appendix E, insofar as it requires a licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency; and (5) plant-specific licensing commitments made in response to NUREG-0737 (Items II.B.2, II.B.3, II.F.1, III.D.1.1, III.A.1.2, and III.D.3.4). Specific review criteria are contained in SRP Sections 15.0.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that the licensee has adequately accounted for the effects of the proposed EPU. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC-19, as well as applicable acceptance criteria denoted in SRP Section 15.0.1. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff has reviewed the alternative source term methodology used by the licensee in evaluating the effects of the proposed EPU and concludes that changes continue to provide a sufficient margin of safety with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and parameter inputs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the implementation of an alternative source term.

[2.9.3 Additional Review Areas (Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

NOTE: Use Sections 2.9.2 - 2.9.8 below if the licensee's radiological consequences analyses are not based on an alternative source term (i.e., if the analyses are based on a traditional source term (i.e., TID-14844)

2.9.2 Radiological Consequences of Control Rod Drop Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a control rod drop accident (CRDA). The NRC staff's review included an examination of (1) the plant's response to the accident, (2) the release of fission products from the core to the environment via the turbine and condensers as a result of the accident, (3) and the calculation of radiological doses at the exclusion area boundary (EAB) and low population zone (LPZ) outer boundary, and in the control room due to the releases from the accident. The NRC's acceptance criteria for the radiological consequences of a control rod drop accident are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.4.9.A, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a control rod drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated control rod drop accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values in 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a control rod drop accident.

2.9.3 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analysis of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary (e.g., instrument lines and sample lines). The NRC staff's review included (1) the identification of small lines postulated to fail and the isolation provisions for these lines; (2) the failure scenario; (3) the models and assumptions for the calculation of the radiological doses for the postulated failure; and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident, and (2) GDC-55, insofar as it establishes isolation requirements for small-diameter lines connected to the primary system that form the basis of meeting 10 CFR 100.11. Specific review criteria are contained in SRP Sections 6.4 and 15.6.2, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs will remain acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are substantially below the exposure guideline values of 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary.

2.9.4 Radiological Consequences of Main Steamline Failure Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of an MSLB accident outside the containment to ensure that radioactive releases due to such an event are adequately limited by the TS limit on primary coolant activity. The NRC staff's review included two cases for the reactor coolant iodine concentration: (1) an MSLB with a preaccident iodine spike and (2) an MSLB with the maximum equilibrium concentration for continued full-power operation. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.6.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB outside containment and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated MSLB outside containment since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 (assuming a preaccident iodine spike) and are a small fraction of the Part 100 values for an MSLB with the primary coolant at the maximum equilibrium concentration for continued full-power operation. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to a postulated failure of an MSLB outside containment.

2.9.5 Radiological Consequences of a Design-Basis Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a design-basis LOCA. This review included a summary review of the doses from the hypothetical design-basis LOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LOCA doses. The NRC staff's review also included (1) the contribution to the dose due to leakage from the main steam isolation valves (MSIVs); (2) the methodology and results of calculations of the radiological consequences resulting from containment and ESF components and MSIV leakage following a hypothetical LOCA; and (3) an assessment of the containment with respect to the assumptions and the input parameters for the dose calculations. The NRC's calculations were based on pertinent information in the [**Updated Safety Analysis Report *or* Updated Final Safety Analysis Report**] and considers the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LOCA are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Section 6.4 and Appendices A, B, and D of SRP Section 15.6.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a design-basis LOCA and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a design-basis LOCA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 and the calculated doses in the control room meet the requirements of GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a design-basis LOCA.

2.9.6 Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a postulated FHA. The purpose of this review was to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review included (1) the sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses; (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses; and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.7.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of FHAs and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of FHAs.

2.9.7 Radiological Consequences of Spent Fuel Cask Drop Accidents

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations. The NRC staff's review was conducted to verify various design and operational aspects of the system. The NRC staff's review included (1) determining a need for a design-basis radiological analysis sequence of events; (2) models and assumptions used by the licensee for the calculation of the radiological doses; (3) comparing calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from spent fuel cask drop accidents, including the effects on control room habitability; and (4) examining the relationship of the operational modes of the standby gas treatment system (SGTS) to the time sequence of the accident in order to give proper credit, in a dual containment design where the fuel building atmosphere may be exhausted through the SGTS. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. Specific review criteria are contained in SRP Sections 6.4 and 15.7.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a spent fuel cask drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to spent fuel cask drop accidents.

[2.9.8 Additional Review Areas (Source Terms and Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]