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Part III

Nuclear Regulatory Commission

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations; Notice

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 2, 2007, to August 15, 2007. The last biweekly notice was published on August 14, 2007 (72 FR 45454).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rulemaking, Directives and Editing Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the basis for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner/requestor to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer of the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, PA

Date of amendment request: June 29, 2007.

Description of amendment request: The proposed license amendment would revise the TMI–1 Technical Specifications 3.3.1.3, 3.3.2.1 and 4.1, to reflect a change to the Reactor Building spray system buffering agent from sodium hydroxide to trisodium phosphate dodecahydrate. This proposed change is designed to minimize the potential for exacerbating sump screen blockage under post loss of coolant event conditions by limiting potential adverse chemical interactions between the buffering agent and certain insulation materials used in the TMI–1 containment.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

For the proposed change, trisodium phosphate dodecahydrate (TSP) will be used as a buffer for post-accident pH control and will replace the existing buffer. The buffer material and means of storage and delivery are not initiators for previously analyzed accidents. The accident mitigation function of the replacement buffer is the same as the existing buffer. The pH of the water in the emergency sump following a loss of coolant accident (LOCA) will be adjusted with TSP rather than sodium hydroxide (NaOH) to be within a range that will reduce the potential for elemental iodine re-evolution and long term stress corrosion during the recirculation mode of emergency core cooling system (ECCS) operation. In addition, the replacement buffer will reduce the formation of precipitates resulting from chemical reactions between the recirculating spray solution and insulating materials in the Reactor Building (RB), thus reducing the potential for ECCS emergency sump intake screen blockage. The proposed sump pH range will not result in an increase in post-LOCA hydrogen generation. The proposed isolation of the sodium hydroxide tank, and the installation of TSP in baskets has been evaluated for impacts on accident effects and the safety functions of required systems, structures, and components (SSCs). The RB emergency sump solution pH profile resulting from the proposed change has been evaluated for impacts on environmental qualification of SSCs. The accident mitigation functions of required SSCs will not be affected by the proposed change.

As a part of the proposed change, the radiological consequences of a postulated LOCA have been reanalyzed using Standard Review Plan (SRP) 6.5.2, ''Containment Spray as a Fission Product Cleanup System,'' and the Alternate Source Term (AST) guidance in Regulatory Guide 1.183. The analysis considered the use of a plain borated water spray during the post-LOCA injection phase and a spray mixture with a minimum pH of 7.3 during the recirculation phase. The results of the reanalysis show that the consequences of the accident are not increased. The calculated doses at the Exclusion Area Boundary, Low Population Zone boundary, and in the Control Room remain within 10 CFR 50.67 AST dose limits.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change will replace the existing spray additive design using sodium hydroxide solution stored in a tank with TSP contained in baskets located on the floor of the RB. The TSP storage and delivery method is passive. The baskets are constructed of stainless steel to resist corrosion and are seismically qualified. The existing sodium hydroxide tank, associated piping, and valves will no longer be used and will be permanently isolated, but their structural integrity will be maintained. The RB spray system will perform the same function and operate in the same manner for the proposed change; however, the sodium hydroxide tank isolation valves will no longer be required to open on an engineered safeguards actuation signal. The accident mitigation function of TSP will be the same as the existing buffer, sodium hydroxide. The TSP will act as a buffering agent to raise the pH of the water in the containment emergency sump to greater than 7.3 for long-term post-LOCA RB spray recirculation. The SSCs required for post-LOCA accident mitigation have been evaluated for the proposed change including the effects of the modified emergency sump solution pH profile. No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change from sodium hydroxide to TSP will not reduce the effectiveness of the post-LOCA pH control buffer. The TSP will buffer the sump water sufficiently to assure that the resulting mixture pH is > 7.3 and < 8.0. This pH level will be effective in reducing the potential for iodine re-evolution during the recirculation phase of a LOCA, preventing long-term stress corrosion cracking of austenitic stainless steel, and minimizing post-LOCA hydrogen generation. In addition, the use of TSP will reduce the formation of precipitates resulting from chemical reactions between the recirculating spray solution and insulating materials in the RB, thus reducing the potential for ECCS emergency sump intake screen blockage. The proposed use of SRP 6.5.2 guidance, which is an NRC-approved methodology, for post-LOCA dose calculations does not result in a reduction in a margin of safety. The proposed change does not adversely affect the performance of SSCs required for post-LOCA mitigation, and does not affect an operating parameter or setpoint used in the accident analyses to establish a margin of safety. Also, the proposed change does not affect a margin of safety associated with containment functional performance.

Therefore, the proposed change does not involve a significant reduction in any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mr. Bradley Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Harold K. Chernoff.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, SC

Date of amendment request: July 17, 2007.

Description of amendment request: A change is proposed to the standard technical specifications (STS) (NUREGs 1430 through 1434) and plant specific technical specifications (TS), to strengthen TS requirements regarding control room envelope (CRE) habitability by changing the action and surveillance requirements associated with the limiting condition for operation operability requirements for the CRE emergency ventilation system, and by adding a new TS administrative controls program on CRE habitability. Accompanying the proposed TS change are appropriate conforming technical changes to the TS Bases. The proposed revision to the Bases also includes editorial and administrative changes to reflect applicable changes to the corresponding STS Bases, which were made to improve clarity, conform with the latest information and references, correct factual errors, and achieve more consistency among the STS NUREGs. The proposed revision to the TS and associated Bases is consistent with STS as revised by TSTF–448, Revision 3.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: Thomas H. Boyce.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, WI

Date of amendment request: June 12, 2007.

Description of amendment request: The proposed amendment would revise the nuclear instrumentation system permissive setpoints in Technical Specification (TS) Table 3.5–2, ''Instrument Operation Conditions for Reactor Trip,'' revise the Table format, and revise TS 2.3, ''Instrumentation System,'' to make consistent with other proposed changes to the TS.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment does not change the probability or consequences of any previously evaluated accidents in the KPS [Kewaunee Power Station] updated safety analysis report (USAR). The proposed amendment would modify the TS setpoint values for the P–7 and P–10 permissives. The actual plant settings will continue to be approximately 10% of rated reactor power. The reactor protection system (RPS) is designed to monitor various plant parameters and initiate a reactor trip in the event these parameters are outside predetermined limits. The RPS is not an accident initiator and therefore, changing the setpoints for these permissives will not increase the probability of an accident previously evaluated.

The proposed amendment would add a setpoint band to the current TS required settings for permissive P–7 and P–10 to accommodate proper setting of the permissives. The only previously evaluated accident that is potentially affected by the proposed changes is the Uncontrolled Rod Cluster Assembly Rod Withdrawal At-Power (RWAP) accident analysis. The effects of these setpoint changes have been evaluated and determined not to have a significant effect on the consequences of the RWAP accident analysis results. The acceptance criteria for the RWAP accident analysis continue to be met. Therefore the proposed changes would not increase the consequences of an accident previously evaluated.

Therefore the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment modifies the TS setpoint values for permissives P–7 and P–10. The actual plant settings will continue to be approximately 10% power. The proposed changes affect the power level at which RPS trip functions are enabled or blocked to ensure proper operation of the RPS. The changes do not add any new systems, structures or components (SSCs) or physically modify any existing SSCs with the possibility of creating a new accident.

The proposed amendment does not functionally affect the operation of any SSC important to safety or its ability to perform its design function. Additionally, the proposed amendment does not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment would add a setpoint band to the current TS required settings for permissivies P–7 and P–10 to accommodate proper setting of the permissives. The safety function of the nuclear instrumentation system and the affected permissives are not affected by this proposed change.

The only safety analysis in the KPS USAR potentially affected by these proposed changes is the Uncontrolled Rod Cluster Assembly Rod Withdrawal At-Power (RWAP) event analysis. Evaluation of the RWAP event analysis results demonstrated that the RWAP would not have a significant effect on a margin of safety.

The effects of the proposed change have been evaluated and all safety analysis acceptance criteria will continue to be met.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Acting Branch Chief: Travis L. Tate.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station (KPS), Kewaunee County, Wisconsin

Date of amendment request: July 2, 2007.

Description of amendment request: The proposed amendment would delete operating license (OL) condition 2.C (5), ''Fuel Burnup,'' which restricts maximum rod average burnup to 60 giga-watt days per metric ton uranium (GWD/MTU). Deletion of the OL condition will provide the opportunity to increase maximum rod average burnup to as high as 62 GWD/MTU and allow fuel management flexibility.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Deletion of KPS OL condition 2.C (5) does not add, delete, or modify any KPS systems, structures, or components (SSCs). The proposed amendment would effectively allow future increases in the KPS maximum rod average burnup limit using currently existing fuel management methods and models that have been reviewed and approved by the NRC [Nuclear Regulatory Commission].

Maximum average rod burnup limits will continue to be maintained within safe and acceptable limits using these fuel management methods and models. Nuclear fuel is the only plant component potentially affected by increasing the maximum rod average burnup limit. Increasing the KPS maximum rod average burnup limit does not affect the thermal hydraulic response or the radiological consequences of any previously evaluated accident. The fuel rod design criteria will continue to be met at the maximum burnup limits allowed under the current fuel management and evaluation processes. An increase to the maximum rod average burnup limit will not increase the likelihood of a malfunction of nuclear fuel since the fuel currently used at KPS has been designed to support a maximum rod average burnup well in excess of 62 GWD/MTU.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment would delete a KPS OL condition that limits maximum rod average burnup. The proposed amendment would effectively allow future increases in the KPS maximum rod average burnup limit using currently existing fuel management methods and models that have been reviewed and approved by the NRC. Nuclear fuel is the only component potentially affected by changes to the maximum rod average burnup limit. The proposed amendment does not change the design function of the nuclear fuel or create any credible new failure mechanisms or malfunctions for nuclear fuel. Fuel rod design criteria will continue to be met at the maximum burnup limits allowed under the fuel management methods and models that have been previously reviewed and approved by the NRC. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment deletes a KPS OL condition that limits maximum rod average burnup. The proposed amendment would effectively allow future increases in the KPS maximum rod average burnup limit using currently existing methods and models that have been reviewed and approved by the NRC. The proposed amendment does not result in altering or exceeding a design basis or safety limit for the plant. All current fuel design criteria will continue to be satisfied, and the safety analysis of record, including evaluations of the radiological consequences of design basis accidents, will remain applicable.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Acting Branch Chief: Travis L. Tate.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, WA

Date of amendment request: July 26, 2007, as superseded by letter dated August 8, 2007.

Description of amendment request: The proposed changes revise the requirements of Technical Specification (TS) 3.3.5.2, ''Reactor Core Isolation Cooling (RCIC) System Instrumentation,'' and TS 3.5.2, ''ECCS [Emergency Core Cooling System]—Shutdown,'' to increase the Condensate Storage Tank (CST) level.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The operation of Columbia in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. Neither of these changes affects the probability of any accident previously evaluated as they do not involve or impact accident initiators.

The proposed change to TS 3.3.5.2 would ensure that the consequences would remain the same as that previously evaluated for during any event in which the RCIC pump was utilized. Adequate volume would be maintained in the CST whenever the RCIC pump was aligned to it to ensure that it did not experience loss of suction due to vortexing.

The proposed changes to TS 3.5.2.2 would ensure that the previously assumed volume of water in the CST would still be available to inject into the reactor vessel during Modes 4 and 5 should the suppression pool not meet minimum volume requirements. Therefore, operation of Columbia in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The operation of Columbia in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change will not create a new or different kind of accident since it only affects the amount of water held in reserve to support reactor vessel inventory loss. The proposed change does not introduce any credible mechanisms for unacceptable radiation release nor does it require physical modification to the plant. The plant has operated well within the existing allowable values. The increased margin provided by the increased level will assure no new or different kinds of accidents result from the proposed change. Therefore, the operation of Columbia in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The operation of Columbia in accordance with the proposed amendment will not involve a significant reduction in the margin of safety. The proposed amendment provides assurance that the RCIC pump suction will be transferred without loss of suction and that 135,000 gallons of CST inventory will continue to be available for injection into the RPV [reactor pressure vessel] under worst case conditions. Therefore, operation of Columbia in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: William A. Horin, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, WA

Date of amendment request: July 30, 2007.

Description of amendment request: The proposed changes revise Technical Specifications (TSs) 1.4, ''Frequency,'' 3.1.5, ''Control Rod Scram Accumulators,'' 3.4.1, ''Recirculation Loops Operating,'' 3.5.1, ''ECCS [Emergency Core Cooling System]—Operating,'' 3.5.2, ''ECCS—Shutdown,'' 3.7.1, ''Standby Service Water (SW) System and Ultimate Heat Sink (UHS),'' 3.8.1, ''AC [Alternating Current] Sources—Operating,'' 3.8.2, ''AC Sources—Shutdown,'' and 5.5.6, ''Inservice Testing Program.'' The proposed changes include updates to adopt approved TS Task Force (TSTF) Travelers 284, Revision 3, ''Add 'Met' vs. 'Perform' to Specification 1.4, Frequency,'' TSTF–479, Revision 0, ''Changes to Reflect Revision of 10 CFR 50.55a,'' and TSTF–485, Revision 0, ''Correct Example 1.4–1.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment is administrative in nature and does not affect analysis inputs or mitigation of analyzed accidents and transients. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. The proposed change does not introduce any new modes of plant operation or make any changes to system setpoints. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed amendment is administrative in nature and does not involve physical changes to plant SSCs [structures, systems, or components], or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The proposed amendment does not involve a change to any safety limit, limiting safety system setting, limiting condition for operation, or design parameters for any SSC. The only minor alteration to the plant design basis is relative to the application of TS 3.4.1. However, as discussed in Section 4 [of the licensee's submittal], this alteration biases the operation of the plant in the direction of safety. The proposed amendment does not impact any safety analysis assumptions and does not involve a change in initial conditions, system response times, or other parameters affecting any accident analysis. For these reasons, the proposed amendment does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: William A. Horin, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, WA

Date of amendment request: July 30, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to establish more effective and appropriate action, surveillance, and administrative TS requirements related to ensuring the habitability of the control room envelope (CRE) in accordance with Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF–448, Revision 3, ''Control Room Habitability.'' Specifically, the proposed amendment would modify TS 3.7.3, ''Control Room Emergency Filtration (CREF) System,'' and add new TS 5.5.14, ''Control Room Envelope Habitability Program,'' to Section 5.5, ''Programs and Manuals.''

The NRC staff issued a ''Notice of Availability of Technical Specification Improvement to Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process'' associated with TSTF–448, Revision 3, in the **Federal Register** on January 17, 2007 (72 FR 2022). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request. In its application dated July 30, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC adopted by the licensee is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE [control room envelope] emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation.

The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the analysis adopted by the licensee and, based upon this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendment involves NSHC.

Attorney for licensee: William A. Horin, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Energy Northwest, Docket No.50–397, Columbia Generating Station, Benton County, WA

Date of amendment request: July 30, 2007.

Description of amendment request: The proposed changes revise Technical Specifications (TSs) 3.3.3.1, ''Post Accident Monitoring (PAM) Instrumentation,'' 3.3.6.1, ''Primary Containment Isolation Instrumentation,'' 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' and 3.6.4.2, ''Secondary Containment Isolation Valves (SCIVs).'' The proposed changes adopt the following TS Task Force (TSTF) Travelers that have been previously approved by the Nuclear Regulatory Commission (NRC): TSTF–45–A, Revision 2, ''Exempt Verification of CIVs [containment isolation valves] that are Not Locked, Sealed or Otherwise Secured,'' TSTF–46–A, Revision 1, ''Clarify the CIV Surveillance to Apply Only to Automatic Isolation Valves,'' TSTF–207–A, Revision 5, ''Completion Time for Restoration of Various Excessive Leakage Rates,'' TSTF–269–A, Revision 2, ''Allow Administrative Means of Position Verification for Locked or Sealed Valves,'' TSTF–295–A, Revision 0, ''Modify Note 2 to Actions of PAM Table to Allow Separate Condition Entry for Each Penetration,'' TSTF–306–A, Revision 2, ''Add Action to LCO [limiting condition for operation] 3.3.6.1 to Give Option to Isolate the Penetration,'' and TSTF–323–A, Revision 0, ''EFCV [excess flow check valve] Completion Time to 72 Hours.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below. The licensee addressed each proposed TSTF separately in its analysis:

TSTF–45–A, Revision 2

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would exempt manual isolation valves and blind flanges located inside and outside the primary containment and in the secondary containment that are locked, sealed, or otherwise secured in position from the periodic verification of valve position required by SRs [surveillance requirements] 3.6.1.3.2 and 3.6.1.3.3, and SR 3.6.4.2.1. The exempted valves are verified to be in the correct position upon being locked, sealed, or secured. Because the valves are in the condition assumed in the accident analysis, the proposed change will not affect the initiators or mitigation of any accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change replaces the periodic verification of valve position with verification of valve position followed by locking, sealing, or otherwise securing the valve in position. Periodic verification is also effective in detecting valve mispositioning. However, verification followed by securing the valve in position is effective in preventing valve mispositioning.

TSTF–46–A, Revision 1

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would revise the verification of PCIV and SCIV closure time to clarify that only power operated, automatic valves are required to be tested. PCIVs and SCIVs are not an initiator of any accident previously evaluated; rather, they serve to mitigate the consequences of evaluated accidents. The proposed change does not change the requirement to verify that power operated, automatic PCIVs and SCIVs close within the time assumed in the accident analysis, but rather, clarifies that non-automatic valves, which the accident analysis does not assume close within a specified time, are not required to be tested to verify the closure time. As a result, the mitigating action of the PCIVs and SCIVs is not affected by this change.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change would revise the verification of PCIV and SCIV closure time to clarify that only power operated, automatic valves are required to be tested, and not all power operated valves. There is no closure time assumed in the accident analysis for power operated PCIVs and SCIVs that are not automatic.

TSTF–207–A, Revision 5

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change revises the Actions of TS 3.6.1.3 to make the presentation consistent with similar Conditions in the ISTS [Improved Standard TSs]. Part of this change would extend the CT [completion time] for hydrostatically tested lines on a closed system to 72 hours for

Condition D. Most of the proposed changes do not affect the requirements in the TS and have no effect on the initiation or mitigation of any accident previously evaluated. Leakage of hydrostatically tested lines on a closed system is not an initiator of any accident previously evaluated. The consequences of a previously evaluated accident during the extended CT are the same as the consequences during the existing CT.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed changes are editorial in nature and do not affect the requirements of the TS. Extension of the CT for hydrostatically tested lines on a closed system to 72 hours does not represent a significant reduction in safety given the reliability of closed systems. Nonetheless, leakage can be isolated restored by isolating the penetration with a valve not exceeding the leakage limits.

TSTF–269–A, Revision 2

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change modifies TS 3.6.1.3 and TS 3.6.4.2. Both TS 3.6.1.3 and TS 3.6.4.2 require penetrations with an inoperable isolation valve to be isolated and periodically verified to be isolated. A Note is added to TS 3.6.1.3, Actions A and C, and TS 3.6.4.2, Action A, to allow isolation devices that are locked, sealed, or otherwise secured to be verified by use of administrative means. The proposed change does not affect any plant equipment, test methods, or plant operation, and are not initiators of any analyzed accident sequence. The inoperable containment penetrations will continue to be isolated, and hence perform their isolation function. Operation in accordance with the proposed TS will ensure that all analyzed accidents will continue to be mitigated as previously analyzed.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change will not affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The PCIVs and SCIVs will continue to be operable or will be isolated as required by the existing specifications.

TSTF–295–A, Revision 0

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change clarifies the separate condition entry Note in TS 3.3.3.1 for Function 7, ''PCIV Position.'' The proposed change does not affect any plant equipment, test methods, or plant operation, and are not initiators of any analyzed accident sequence. The actions taken for inoperable PAM channels are not changed. Operation in accordance with the proposed TS will ensure that all analyzed accidents will continue to be mitigated as previously analyzed.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change will not affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The PAM channels will continue to be operable or the existing, appropriate actions will be followed.

TSTF–306–A, Revision 2

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change revises TS 3.3.6.1 by adding an Actions Note that would allow penetration flow paths to be unisolated intermittently under administrative controls. Furthermore, the TIP [traversing incore probe] isolation system is segregated into a separate Function, allowing 24 hours to isolate the penetration. The proposed change does not affect any plant equipment, test methods, or plant operation, and are not initiators of any analyzed accident sequence. The allowance to unisolate a penetration flow path will not have a significant effect on the mitigation of any accident previously evaluated because the penetration flow path can be isolated, if needed, by a dedicated operator. The option to isolate a TIP penetration will ensure the penetration will perform as assumed in the accident analysis. Operation in accordance with the proposed TS will ensure that all analyzed accidents will continue to be mitigated as previously analyzed.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change will not affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The allowance to unisolate a penetration flow path will not have a significant effect on a margin of safety because the penetration flow path can be isolated manually, if needed. The option to isolate a TIP penetration will ensure the penetration will perform as assumed in the accident analysis.

TSTF–323–A, Revision 0

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would revise Action C of TS 3.6.1.3 to provide a 72-hour CT instead of a 12 hour CT to isolate an inoperable EFCV. PCIVs are not an initiator of any accident previously evaluated. The consequences of a previously evaluated accident during the extended CT are the same as the consequences during the existing CT.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The PCIVs serve to mitigate the potential for radioactive release from the primary containment following an accident. The design and response of the PCIVs to an accident are not affected by this change. The revised CT is appropriate given the EFCVs are on penetrations that have been found to have acceptable barrier(s) in the event that the single isolation valve failed.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: William A. Horin, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Entergy Nuclear Operations, Inc., Docket No. 50–255, Palisades Nuclear Plant, Van Buren County, MI

Date of amendment request: May 22, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 5.5.7, ''Inservice Testing Program'' to: (1) Delete reference to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI and incorporate reference to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code), and (2) address the applicability of Surveillance Requirement (SR) 3.0.2 to other normal and accelerated frequencies specified as two years or less in the inservice testing (IST) program.

The proposed amendment incorporates changes based on U.S. Nuclear Regulatory Commission (NRC)—approved Technical Specification Task Force (TSTF) TSTF–479–A, ''Changes to Reflect Revision of 10 CFR 50.55a,'' Revision 0, as modified by NRC-approved TSTF–497, ''Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of Two Years or Less,'' Revision 0. The proposed changes include two deviations from the NRC-approved TSTFs that are administrative in nature: (1) Addition of ''ASME'' to TS 5.5.7 to make references to ''ASME OM Code'' and (2) use of the term ''intervals'' instead of ''frequencies.'' Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes do not have any impact on the integrity of any plant system, structure, or component that initiates an analyzed event. The proposed changes would not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Thus, the probability of any accident previously evaluated is not significantly increased.

The proposed changes do not affect the ability to mitigate previously evaluated accidents, and do not affect radiological assumptions used in the evaluations. The proposed changes do not change or alter the design criteria for the systems or components used to mitigate the consequences of any design basis accident. The proposed amendment does not involve operation of the required structures, systems, or components (SSCs) in a manner or configuration different from those previously recognized or evaluated. Thus, the radiological consequences of any accident previously evaluated are not increased.

Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment does not involve a physical alteration of any SSC or a change in the way any SSC is operated. The proposed amendment does not involve operation of any required SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms would be introduced by the changes being requested.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The amendment does not involve a significant reduction in a margin of safety. The proposed amendment does not affect the acceptance criteria for any safety analysis analyzed accidents or anticipated operational occurrences. The proposed amendment does not alter the limiting values and acceptance criteria used to judge the continued acceptability of components tested by the IST Program. The safety function of the affected pumps and valves will be maintained.

Therefore, the proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mr. William Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Ave., White Plains, NY 10601.

NRC Acting Branch Chief: Travis L. Tate.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, AR

Date of amendment request: July 31, 2007.

Description of amendment request: The proposed amendment will revise Arkansas Nuclear One, Unit 2 (ANO–2) Technical Specification (TS) 6.6.5, Core Operating Limits. The proposed change will add new analytical methods to support the implementation of Next Generation Fuel (NGF).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the COLR [Core Operating Limits Report] TS are administrative in nature and have no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Changes to the calculated core operating limits may only be made using NRC-approved methodologies, must be consistent with all applicable safety analysis limits, and are controlled by the 10 CFR 50.59 process.

The proposed change will add the following topical reports to the list of referenced core operating analytical methods.

WCAP–16500–P and Final Safety Evaluation (SE)

Westinghouse topical report WCAP–16500–P describes the methods and models that will be used to evaluate the acceptability of CE [Combustion Engineering] 16 x 16 NGF at CE plants. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF, the new core design will be analyzed with applicable NRC staff-approved codes and methods.

WCAP–12610–P–A and CENPD–404–P–A Addendum 1–A

The proposed change allows the use of methods required for the implementation of Optimized ZIRLO™ clad fuel rods. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

WCAP–16523–P and Final Safety Evaluation

This topical report describes the departure from nucleate boiling [DNB] correlations that will be used to account for the impact of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF, the new core design will be analyzed with applicable NRC staff-approved codes and methods.

CENPD–387–P–A

The proposed addition of this topical report provides the [DNB] correlation that will be used to evaluate the DNB impact of non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

CENPD–132, Supplement 4–P–A, Addendum 1–P and Final Safety Evaluation

The addendum provides an optional steam cooling model that can be used for Emergency Core Cooling System (ECCS) Performance analyses to support the implementation of the CE 16 x 16 NGF fuel assembly design. The optional steam cooling model is not being used to support implementation of CE 16 x 16 NGF assemblies in ANO–2 at this time. However, Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

Assumptions used for accident initiators and/or safety analysis acceptance criteria are not altered by the addition of these topical reports.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change identifies changes in the codes used to confirm the values of selected cycle-specific reactor physics parameter limits. The proposed change allows the use of methods required for the implementation of CE 16 x 16 NGF. The proposed addition of the referenced topical reports has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. These changes are administrative in nature and do not result in a change to the physical plant or to the modes of operation defined in the facility license.

WCAP–16500–P and Final Safety Evaluation

The proposed change adds Westinghouse topical report WCAP–16500–P, which describes the methods and models that will be used to evaluate the acceptability of CE 16 x 16 NGF at CE plants. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF, the new core design will be analyzed with applicable NRC staff-approved codes and methods.

WCAP–12610–P–A and CENPD–404–P–A Addendum 1–A

The proposed change allows the use of methods required for the implementation of Optimized ZIRLO™ clad fuel rods. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

WCAP–16523–P and Final Safety Evaluation

This topical report describes the [DNB] correlations that will be used to account for the impact of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the SE will be met.

CENPD–387–P–A

The proposed addition of this topical report provides the [DNB] correlation that will be used to evaluate the DNB impact of non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

CENPD–132, Supplement 4–P–A, Addendum 1–P and Final Safety Evaluation

The addendum provides an optional steam cooling model that can be used for ECCS Performance analyses to support the implementation of the CE 16 x 16 NGF fuel assembly design. The optional steam cooling model is not being used to support implementation of CE 16 x 16 NGF assemblies in ANO–2 at this time. However, Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not amend the cycle-specific parameter limits located in the COLR from the values presently required by the TS. The individual specifications continue to require operation of the plant within the bounds of the limits specified in COLR.

The addition of the following topical reports to the list of analytical methods referenced in the COLR is administrative in nature:

□ WCAP–16500–P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP–16500–P, Revision 0, ''CE [Combustion Engineering] 16 x 16 Next Generation Fuel [(NGF)] Core Reference Report''

□ WCAP–12610–P–A and CENPD–404–P–A Addendum 1–A

□ WCAP–16523–P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR), WCAP–16523–P, ''Westinghouse Correlations WSSV and WSSV–T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes''

□ CENPD–387–P–A

□ CENPD–132, Supplement 4–P–A, Addendum 1–P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD–132 Supplement 4–P–A, Addendum 1–P, ''Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model—Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood''

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Terence A. Burke, Associate General Council—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Thomas G. Hiltz.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3,York and Lancaster Counties, PA

Date of application for amendments: November 17, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement 3.3.1.1.8 to increase the frequency interval between Local Power Range Monitor (LPRM) calibrations from 1000 megawatt days per ton (MWD/T) average core exposure to 2000 MWD/T average core exposure. The LPRM system provides signals to associated nuclear instrumentation systems that serve to detect conditions in the core that have the potential to threaten the overall integrity of the fuel barrier. The LPRM system also incorporates features designed to diagnose and display various system trip and inoperative conditions.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed amendment revises the surveillance interval for the LPRM calibration from 1000 MWD/T average core exposure to 2000 MWD/T average core exposure. Increasing the frequency interval between required LPRM calibrations is acceptable due to improvements in core monitoring processes and nuclear instrumentation and therefore, the revised surveillance interval continues to ensure that the LPRM detector signal is adequately calibrated.

This change will not alter the operation of process variables, structures, systems, or components as described in the PBAPS Updated Final Safety Analysis Report (UFSAR). The proposed change does not alter the initiation conditions or operational parameters for the LPRM system and there is no new equipment introduced by the extension of the LPRM calibration interval. The performance of the APRM, OPRM and RBM systems is not significantly affected by the proposed surveillance interval increase. As such, the probability of occurrence of a previously evaluated accident is not increased.

The radiological consequences of an accident can be affected by the thermal limits existing at the time of the postulated accident; however, LPRM chamber exposure has no significant effect on the calculated thermal limits since LPRM accuracy does not significantly deviate with exposure. For the LPRM extended calibration interval, the total nodal power uncertainty remains less than the uncertainty assumed in the thermal analysis basis safety limit, maintaining the accuracy of the thermal limit calculation. Therefore, the thermal limit calculation is not significantly affected by LPRM calibration frequency, and thus the radiological consequences of any accident previously evaluated are not increased.

Therefore, based on the above information, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The performance of the APRM, OPRM and RBM systems is not significantly affected by the proposed LPRM surveillance interval increase. The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not change or introduce any new equipment, modes of system operation or failure mechanisms.

Therefore, based on the above information, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed change has no impact on equipment design or fundamental operation, and there are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety as a result of the proposed LPRM surveillance interval increase. The performance of the APRM, OPRM and RBM systems is not significantly affected by the proposed change. The margin of safety can be affected by the thermal limits existing at the time of the postulated accident; however, uncertainties associated with LPRM chamber exposure have no significant effect on the calculated thermal limits. The thermal limit calculation is not significantly affected since LPRM sensitivity with exposure is well defined. LPRM accuracy remains within the total nodal power uncertainty assumed in the thermal analysis basis; thereby maintaining thermal limits and the safety margin. The proposed change does not affect safety analysis assumptions or initial conditions and therefore, the margin of safety in the original safety analyses are maintained.

Therefore, based on the above information, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for Licensee: Mr. J. Bradley Fewell, Associate General Counsel, Exelon Generation Company LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Harold K. Chernoff.

Florida Power and Light Company (FPL), Docket Nos. 50–335 and 50–389, St. Lucie Plant, Units 1 and 2, St. Lucie County, FL

Date of amendment request: July 16, 2007.

Description of amendment request: The proposed amendment would modify the technical specification (TS) requirements related to control room envelope (CRE) habitability in accordance with Technical Specification Task Force (TSTF) Traveler TSTF–448, Revision 3, ''Control Room Habitability,'' published in the **Federal Register** on January 17, 2007 (Volume 72, Number 10), as part of the consolidated line item improvement process. Specifically by modifying Unit 1 TS 3.7.7.1, ''Control Room Emergency Ventilation System (CREVS),'' and Unit 2 TS 3.7.7,'' Control Room Emergency Air Cleanup System (CREACS),'' and adding a new Unit 1 and Unit 2 TS Section 6.8.4.m.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Branch Chief: Thomas H. Boyce.

Florida Power and Light Company (FPL), Docket Nos. 50–335, St. Lucie Plant, Unit 1, St. Lucie County, FL

Date of amendment request: July 16, 2007.

Description of amendment request: The proposed amendment would modify the facilities operating licensing bases to adopt the alternative source term (AST) as allowed in 10 CFR 50.67 and described in Regulatory Guide (RG) 1.183. The licensee proposes to revise the plant licensing basis through reanalysis of the following radiological consequences of the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents: Loss-of-Coolant Accident, Fuel Handling Accident, Main Steam Line Break, Steam Generator Tube Rupture, Reactor Coolant Pump Shaft Seizure, Control Element Assembly Ejection, and Inadvertent Opening of a Main Steam Safety Valve.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Alternative source term calculations have been performed for St. Lucie Unit No. 1 which demonstrate that the dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed changes do not modify the design or operation of the plant. The use of the AST only changes the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of any accident. The AST has been utilized in the analysis of the limiting design basis accidents listed above. The results of the analyses, which include the proposed changes to the Technical Specifications [TSs], demonstrate that the dose consequences of these limiting events are all within the regulatory limits.

With the exception of the deletion of SRs 4.6.6.1.c.[3].b and 4.7.8.1.c.[3].b, the proposed Technical Specification changes are consistent with, or more restrictive than, the current TS requirements. The proposed filter testing requirements continue to ensure that the associated filtration systems function as described in the UFSAR and as assumed in the accident analyses. None of the affected systems, components or programs are related to accident initiators.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not affect any plant structures, systems, or components. The operation of plant systems and equipment will not be affected by this proposed change. Neither implementation of the alternative source term methodology, establishing more restrictive TS requirements, nor deleting SRs 4.6.6.1.c.[3].b and 4.7.8.1.c.[3].b have the capability to introduce any new failure mechanisms or cause any analyzed accident to progress in a different manner.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

The proposed implementation of the alternative source term methodology is consistent with NRC Regulatory Guide 1.183. With the exception of the deletion of SRs 4.6.6.1.c.[3].b and 4.7.8.1.c.[3].b, the proposed Technical Specification changes are consistent with, or more restrictive than, the current TS requirements. The proposed TS requirements support the AST revisions to the limiting design basis accidents. The proposed filter testing requirements continue to ensure that the associated filtration systems function as described in the UFSAR and as assumed in the accident analyses. As such, the current plant margin of safety is preserved. Conservative methodologies, per the guidance of RG 1.183, have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with use of the alternative source term methodology.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries and in the Control Room are within the corresponding regulatory limits of RG 1.183 and 10 CFR 50.67. The margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are set at or below the 10 CFR 50.67 limits. An acceptable margin of safety is inherent in these limits.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Branch Chief: Thomas H. Boyce.

Florida Power and Light Company (FPL), Docket No. 50–389, St. Lucie Plant, Unit 2, St. Lucie County, FL

Date of amendment request: July 16, 2007.

Description of amendment request: The proposed amendment would modify the facilities operating licensing bases to adopt the alternative source term (AST) as allowed in 10 CFR 50.67 and described in Regulatory Guide (RG) 1.183. The licensee proposes to revise the plant licensing basis through reanalysis of the following radiological consequences of the Updated Final Safety Analysis Report Chapter 15 accidents: Loss-of-Coolant Accident, Fuel Handling Accident, Main Steam Line Break, Steam Generator Tube Rupture, Reactor Coolant Pump Shaft Seizure, Control Element Assembly Ejection, Letdown Line Break, and Feedwater Line Break.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Alternative source term calculations have been performed for St. Lucie Unit No. 2 which demonstrate that the dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed changes do not modify the design or operation of the plant. The use of the AST only changes the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of any accident. The AST has been utilized in the analysis of the limiting design basis accidents listed above. The results of the analyses, which include the proposed changes to the Technical Specifications [TSs], demonstrate that the dose consequences of these limiting events are all within the regulatory limits.

The proposed Technical Specification Changes are consistent with, or more restrictive than, the current TS requirements. None of the affected systems, components or programs are related to accident initiators.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not affect any plant structures, systems, or components. The operation of plant systems and equipment will not be affected by this proposed change. Neither implementation of the alternative source term methodology nor establishing more restrictive TS requirements have the capability to introduce any new failure mechanisms or cause any analyzed accident to progress in a different manner.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

The proposed implementation of the alternative source term methodology is consistent with NRC Regulatory Guide 1.183. The proposed Technical Specification changes are consistent with, or more restrictive than, the current TS requirements. These TS requirements support the AST revisions to the limiting design basis accidents. As such, the current plant margin of safety is preserved. Conservative methodologies, per the guidance of RG 1.183, have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with use of the alternative source term methodology.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries and in the Control Room are within the corresponding regulatory limits of RG 1.183 and 10 CFR 50.67. The margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are set at or below the 10 CFR 50.67 limits. An acceptable margin of safety is inherent in these limits.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above discussion, FP&L has determined that the proposed change does not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Branch Chief: Thomas H. Boyce.

Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, MN

Date of amendment request: July 3, 2007.

Description of amendment request: The proposed amendments would revise the Technical Specifications

(TSs) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2 to:

1. Revise TS 1.4, ''Frequency'' to modify the second paragraph of Example 1.4–1 to be consistent with the requirements of Surveillance Requirement (SR) 3.0.4 and incorporate the changes in Technical Specification Task Force (TSTF) industry traveler TSTF–485, ''Correct Example 1.4–1.''

2. Revise TS 5.5.7.a, to modify references to Section XI of the American Society of Mechanical Engineers (ASME) Code with references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code), to be consistent with TSTF–479, ''Changes to Reflect Revision of 10 CFR [Code of Federal Regulations] 50.55a.

3. Revise TS 5.5.7.b, to restrict extension of Frequencies to those Frequencies specified as 2 years or less, and take exception to the limitation in SR 3.0.2 which does not apply the 1.25 times extension to Frequencies of 24 months, to be consistent with TSTF–479 and TSTF–497, ''Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less.''

4. Revise TS 5.5.7.d, to modify the referenced ASME Code to be consistent with TSTF–479.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

TSTF–479

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises the Improved Standard Technical Specification (ISTS) Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change revises the improved Standard Technical Specification (ISTS) Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises the improved Standard Technical Specification (ISTS) Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The safety function of the affected pumps and valves will be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TSTF–485

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises Section 1.4, Frequency, Example 1.4–1, to be consistent with Surveillance Requirement (SR) 3.0.4 and Limiting Condition for Operation (LCO) 3.0.4. This change is considered administrative in that it modifies the example to demonstrate the proper application of SR 3.0.4 and LCO 3.0.4. The requirements of SR 3.0.4 and LCO 3.0.4 are clear and are clearly explained in the associated Bases. As a result, modifying the example will not result in a change in usage of the Technical Specifications (TS). The proposed change does not adversely affect accident initiators or precursors, the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Therefore, this change is considered administrative and will have no effect on the probability or consequences of any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

No new or different accidents result from utilizing the proposed change. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change is administrative and will have no effect on the application of the Technical Specification requirements. Therefore, the margin of safety provided by the Technical Specification requirements is unchanged. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

TSTF–497

This Traveler is considered an administrative change to the ISTS NUREGs. Therefore, a regulatory analysis is not provided.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Acting Branch Chief: Travis L. Tate.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Rivers, Manitowoc County, WI

Date of amendment request: July 12, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.6.3, ''Containment Isolation Valves.'' The revision would delete Surveillance

Requirement (SR) 3.6.3.1, which is no longer required due to the containment purge supply and exhaust valve isolation function being replaced with blind flanges. The proposed amendment would also support a change to the Final Safety Analysis Report (FSAR) to revise the requirement to leak check the purge supply and exhaust valves.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change to the containment purge supply penetration and the containment exhaust penetration presents no change in the probability or the consequence of an accident. The penetrations continue to conform to the TS requirements for containment and will be appropriately tested as required by 10 CFR 50 Appendix J. The blind flanges are passive devices not susceptible to an active failure or malfunction that could result in a loss of isolation or leakage that exceeds the limits assumed in the safety analyses. The blind flanges are leak rate tested in accordance with the containment leakage rate testing program. Containment isolation is not lessened by this change.

The change to the containment purge system does not affect the design basis limit for any fission product barrier.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change to the containment purge supply penetration and the containment exhaust penetration does not change the function of the system and does not alter containment isolation. The penetrations continue to conform to the TS requirements for containment isolation and will be appropriately tested as required by 10 CFR 50 Appendix J. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

3. Does the proposed change involve a significant reduction in the margin of safety?

Response: No.

The proposed change will not alter any assumptions, initial conditions or results specified in any accident analysis. The containment purge supply and exhaust penetrations will continue to conform to the TS requirements for containment and will be appropriately tested as required by 10 CFR 50 Appendix J. The blind flanges are passive devices not susceptible to an active failure or malfunction that could result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. The blind flanges are leak rate tested in accordance with the containment leakage rate testing program. Containment isolation is not lessened by this change. Therefore, there is no reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Acting Branch Chief: Travis L. Tate.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, NE

Date of amendment request: July 30, 2007.

Description of amendment request: The proposed amendment by Omaha Public Power District requests changes to the Fort Calhoun Station Unit No.1 Operating License No. DPR–40 to modify the containment spray system actuation logic to preclude automatic start of the containment spray pumps for a loss-of-coolant accident.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The containment spray (CS) system and the containment air cooling and filtering system (CACFS) are not initiators of any accident previously evaluated at the Fort Calhoun Station (FCS). Both systems are accident mitigation systems. Their licensing basis functions are to limit the containment pressure rise and reduce the leakage of airborne radioactivity from the containment by providing a means for cooling the containment following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) inside containment. The proposed modification to the CS system logic shifts the function of containment pressure and temperature control during a LOCA from the [CS] system to the equally capable and reliable containment air coolers. The change in the CS actuation logic does not impact the containment response to the MSLB analysis of record (AOR). The CACFS provides the design heat removal capabilities for the containment during the postulated LOCA. The system is operated to remove atmospheric heat loads from the containment during normal plant operation. Since system components are only lightly loaded during normal operation, system availability and reliability are enhanced. In the unlikely event that normal power sources are lost and one emergency diesel generator fails to operate, one containment air cooling and filtering unit and one containment air cooling unit will operate.

The component cooling water (CCW) system, on which the CACFS is dependent, has sufficient capacity for all normal and shutdown operating modes. In addition, the system is capable of satisfying the design criteria under post design-basis accident (DBA) conditions with the single failure of an active component and a loss of instrument air. Analyses demonstrate that CCW flowrates to essential equipment would be adequate for removing post accident design-basis heat loads.

Following implementation of the proposed change, at least one train of containment air coolers will be available to mitigate a LOCA. Analyses show that one train of coolers can maintain the containment pressure and temperature below the design values; therefore, the proposed change will have no adverse effect on the containment pressure analysis following a LOCA.

Analyses have also shown that one train of containment high-efficiency particulate air (HEPA) filters maintains the radiological consequences doses within regulatory limits; therefore, the proposed change will have no adverse effect on the radiological consequences following a LOCA.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The CACFS was designed to remove heat released to containment atmosphere during the [DBA] to the extent necessary to maintain the structure below the design pressure. The proposed modification to the CS system logic shifts the function of containment pressure and temperature control from the [CS] system to the equally capable and reliable containment air coolers. The use of CACFS, as a means of containment pressure control, has been evaluated for the LOCA event and found to result in an acceptable peak containment pressure (peak pressure less than 60 psig [pounds per square inch gauge]). Radiological consequences were evaluated for the use of CACFS in this application using the guidance provided in Regulatory Guide (RG) 1.183. This radiological analysis demonstrates that the dose consequences are in compliance with applicable regulatory requirements. The estimated dose consequences at the exclusion area boundary (EAB), low population zone (LPZ), and control room (CR) remain within the acceptance criteria of 10 CFR 50.67 as supplemented by RG 1.183 and the standard review plan (SRP) 15.0.1. The assessment also demonstrates that the dose consequences in the technical support center (TSC) remain compliant with regulatory guidance provided in Supplement 1 of NUREG–0737.

No credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis have been created and none of the initial condition assumptions of any accident evaluated in the safety analysis are impacted.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The containment building and associated penetrations are designed to withstand an internal pressure of 60 psig at 305 °F [degrees Fahrenheit], including all thermal loads resulting from the temperature associated with this pressure, with a leakage rate of 0.1 percent by weight or less of the contained volume per 24 hours. The containment air coolers are credited for maintaining containment pressure and temperatures within design limitations, and assure that the release of fission products to the environment following a [DBA] will not exceed regulatory guidelines for a large break (LB) LOCA.

The [CS] system and containment air coolers continue to be credited for limiting peak containment pressure for an MSLB.

Adequate NPSH [net positive suction head] margin is maintained for the HPSI [high-pressure safety injection] pumps during the recirculation phase of a[n] LBLOCA due to the reduction in ECCS [emergency core cooling system] sump strainer pressure drop.

The CACFS operates independently of the CS system to remove heat from the containment atmosphere. The CACFS consists of two redundant trains, each train with one air cooling and filtering unit and one air cooling unit, for a total of four cooling units. Operation of the CACFS, in accordance with analyses completed for the 2006 steam generator replacement, is and will continue to be credited in the MSLB containment pressure analysis. The operation and maintenance of the CACFS are not impacted by this proposed change. Therefore, the containment heat removal licensing basis is not adversely affected by the proposed change. The ability to maintain containment peak pressure and temperature, as well as long-term containment pressure and temperature, is maintained.

The LBLOCA 10 CFR 50.46 analysis assumes that there will be three CS pumps operating when evaluating the effects of containment pressure on ECCS performance. This assumption minimizes containment pressure, to conservatively evaluate ECCS performance in response to a LOCA. Eliminating operation of the CS pumps improves ECCS performance and thus increases margin to 10 CFR 50.46 limits on peak clad temperature, therefore, the existing analysis remains bounding as is.

In summary, following implementation of the proposed change:

□ Peak containment pressure for analyzed DBAs remains within design limits;

□ Radiological releases remain within the limits of 10 CFR 50.67; and

□ The currently calculated peak clad temperature following a LOCA remains bounding.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, NE

Date of amendment request: July 31, 2007.

Description of amendment request: The proposed amendment will modify Technical Specification (TS) requirements to support a planned inverter modification to be installed during the 2008 refueling outage. The inverter modification will require revisions to TS 2.7(1), 2.7(2), and 3.7(5), and the associated Bases sections to allow for the addition of two safety-related swing inverters.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The addition of two safety-related swing inverters to the 120 V a-c [Volts alternating current] vital instrument buses is not an initiator of any previously evaluated accidents. The swing inverters will not prevent safety systems from performing any of the accident mitigation functions assumed in the safety analysis. The revisions proposed for the Technical Specifications (TS) take advantage of the operational flexibility provided by the swing inverters yet maintain current TS requirements that four inverters be operable.

Similarly, the change maintains the current TS allowance for one of the required inverters to be inoperable for up to twenty-four hours provided all current TS requirements for operability are met.

Although continued operation for up to twenty-four hours with one of the required inverters inoperable is allowed, the addition of the two safety-related swing inverters is expected to decrease the amount of time that the station must operate with less than four inverters. This is because the design allows the inoperable inverter to be replaced by its associated swing (or non-swing) inverter. Reducing the need to shut the station down due to an inoperable inverter also reduces the risk associated with mode transition to shutdown.

The correction of two typographical errors and correcting spacing inconsistencies in the text are administrative changes that do not involve a significant increase in the probability or consequences of any accident previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The design function of the safety-related inverters is unchanged. The addition of the safety-related swing inverters and their bypass sources to the 120 volt a-c vital instrument distribution system allows preventative maintenance, repair and for testing to be performed online. If a safety-related inverter becomes inoperable or is otherwise out-of-service, its instrument bus is manually transferred to the associated swing inverter. If a required inverter should fail, the time that the station will operate with less than the four inverters required by TS 2.7(1)j should, in most cases, be less due to the ability to place an associated inverter online. Reducing the need to shut the station down due to an inoperable inverter also reduces the risk associated with mode transition to shutdown.

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

3. Does the proposed amendment involve a significant reduction in the margin of safety?

Response: No

The design function of the safety-related inverters is unchanged. The addition of the safety-related swing inverters to the 120 volt a-c vital instrument distribution system allows preventative maintenance or repair of a safety-related inverter to be performed online since its instrument bus can be manually transferred to the associated swing inverter. Installation of the safety-related swing inverters does not require changes to accident analyses or results. The revisions proposed for the TS maintain current TS requirements that four inverters be operable. Should a required inverter fail, the time that the station will operate with less than the four inverters required by TS 2.7(1)j should, in most cases, be less due to the ability to place an associated inverter online. Reducing the need to shut the station down due to an inoperable inverter also reduces the risk associated with mode transition to shutdown. In addition, administrative controls are in place to ensure the current station battery capacity is not degraded and to ensure battery margin is adequately maintained as a result of the inverter modification.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, TN

Date of amendment request: July 26, 2007.

Description of amendment request: The proposed amendment would add a new reference to Technical Specification 6.9.1.14.a, which lists documents that have been approved by the U.S. Nuclear Regulatory Commission for use in determining the core operating limits. The new reference is the Areva NP, Inc. topical report EMF–2103P–A, ''Realistic Large Break LOCA Methodology for Pressurized Water Reactors.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change adds an approved analytical method for evaluating large break loss of coolant accidents (LOCAs). The proposed change will not affect previously evaluated accidents because they continue to be analyzed by NRC approved methodologies to ensure required safety limits are maintained. The acceptance criteria of the SQN Final Safety Analysis Report analyzed accidents and anticipated operational occurrences are not affected by the proposed addition of the realistic large break LOCA methodology. As the evaluations for accidents and operation occurrences are not adversely affected, the proposed change will not increase the consequences of a postulated event. The proposed change does not result in any modification of the plant equipment or operating practices and therefore, does not alter plant conditions or plant response prior to or after postulated events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

As previously noted, the proposed change does not result in any modification of the plant equipment or operating practices and therefore, does not alter plant conditions or plant response prior to or after postulated events. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change does not alter plant equipment including the automatic accident mitigation setpoints designed to mitigate the affects of a postulated accident. The accident analyses and plant safety limits continue to be acceptable as evaluated by NRC approved methodologies. The proposed application of the realistic large break LOCA methodology ensures acceptable margins and limits for fuel core designs. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Thomas H. Boyce.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

AmerGen Energy Company, LLC, Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, NJ

Date of application for amendment: November 27, 2006.

Brief description of amendment: The amendment revised the required submittal date for the Annual Radioactive Effluent Release Report. Specifically, the required submittal date is revised from ''within 60 days after January 1, each year,'' to ''prior to May 1 of each year.''

Date of Issuance: August 8, 2007.

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No: 264.

Facility Operating License No. DPR–16: The amendment revised the license and the Technical Specifications.

Date of initial notice in **Federal Register**: May 8, 2007 (72 FR 26174).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 8, 2007.

No significant hazards consideration comments received: No.

Detroit Edison Company, Docket No. 50–341, Fermi, Unit 2, Monroe County, MI

Date of application for amendment: July 12, 2006, as supplemented by letters dated April 25, May 23, June 15, June 20, and June 29, 2007.

Brief description of amendment: The amendment modifies Conditions, Required Actions and Completion Times in Technical Specification (Ts) 3.8.1, ''AC Sources-Operating,'' associated with the Required Actions when emergency diesel generators are declared inoperable.

Date of issuance: August 1, 2007.

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment No: 175.

Facility Operating License No. NPF–43: Amendment revised the TSs and License.

Date of initial notice in **Federal Register**: August 29, 2006 (71 FR 51225). The April 25, May 23, June 15, June 20, and June 29, 2007, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 2007.

No significant hazards consideration comments received: No.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, MI

Date of application for amendment: January 26, 2007.

Brief description of amendment: The amendment adds a Limiting Condition for Operation (LCO) 3.0.9 to the Technical Specifications (TS), allowing a delay time for entering a supported system TS, when the inoperability is due solely to an unavailable barrier, if risk is assessed and managed. Additionally, the amendment makes editorial changes to LCO 3.0.8 to be consistent with terminology of LCO 3.0.9.

Date of issuance: August 1, 2007.

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment No.: 176.

Facility Operating License No. NPF–43: Amendment revised the TS and License.

Date of initial notice in **Federal Register:** April 10, 2007 (72 FR 17945).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 2007.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, OH

Date of application for amendment: May 30, 2006, as supplemented by letters dated April 24, 2007, and June 27, 2007.

Brief description of amendment: This amendment revises the existing SG tube surveillance program to be consistent with the Nuclear Regulatory Commission's approved TS Task Force (TSTF) Standard TS Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on May 6, 2005 (70 FR 24126). The amendment is also the modification of the SG portion of the TSs requested in NRC Generic Letter (GL) 2006–01, ''Steam Generator Tube Integrity and Associated Technical Specification.''

Date of issuance: July 31, 2007.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 276.

Facility Operating License No. NPF–3: The amendment revised the Technical Specifications and License.

Date of initial notice in **Federal Register**: October 10, 2006 (71 FR 59531). The April 24, 2007, and June 27, 2007 supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 31, 2007.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit No. 2, Oswego County, NY

Date of application for amendment: March 8, 2007.

Brief description of amendment: The amendment revises the Technical Specification requirements for inoperable snubbers by adding Limiting Condition for Operation 3.0.8 using the Consolidated Line Item Improvement Process.

Date of issuance: July 30, 2007.

Effective date: As of the date of issuance to be implemented within 180 days.

Amendment No.: 118.

Renewed Facility Operating License No. NPF–69: Amendment revises the License and Technical Specifications.

Date of initial notice in **Federal Register**: April 24, 2007 (72 FR 20384).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 2007.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–259 Browns Ferry Nuclear Plant, Unit 1, Limestone County, AL

Date of application for amendment: June 25, 2007, as supplemented by letters dated July 3 and 26, 2007 (TS–461).

Brief description of amendment: The amendment deletes License Condition 2.G.(2) as the result of completion of power uprate large transient testing.

Date of issuance: August 14, 2007.

Effective date: The date of issuance, to be implemented within 30 days.

Amendment No.: 272.

Renewed Facility Operating License No. DPR–33: Amendment revised the renewed operating license.

Date of initial notice in **Federal Register**: July 13, 2007 (72 FR 38627). The July 3 and 26, 2007, supplemental letters provided clarifying information that did not expand the scope of the application or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 14, 2007.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application for the amendment complies with the standards and requirements of the

Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397 4209, (301) 415–4737 or by email to *pdr@nrc.gov.*

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also identify the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.[1]

[1] To the extent that the applications contain attachments and supporting documents that are not

Continued

Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Each contention shall be given a separate numeric or alpha designation within one of the following groups:

1. Technical—primarily concerns/ issues relating to technical and/or health and safety matters discussed or referenced in the applications.

2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.

3. Miscellaneous—does not fall into one of the categories outlined above.

As specified in 10 CFR 2.309, if two or more petitioners/requestors seek to co-sponsor a contention, the petitioners/ requestors shall jointly designate a representative who shall have the authority to act for the petitioners/ requestors with respect to that contention. If a petitioner/requestor seeks to adopt the contention of another sponsoring petitioner/requestor, the petitioner/requestor who seeks to adopt the contention must either agree that the sponsoring petitioner/requestor shall act as the representative with respect to that contention, or jointly designate with the sponsoring petitioner/requestor a representative who shall have the authority to act for the petitioners/ requestors with respect to that contention.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by email to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer or the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel and discuss the need for a protective order.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, IL

Date of amendment request: June 29, 2007, as supplemented by letters dated August 1, 2007 and August 2, 2007.

Description of amendment request: The amendments revised the maximum allowed Technical Specification (TS) temperature limit, contained in TS Surveillance Requirement 3.7.3.1, of the cooling water supplied to the plant from the Core Standby Cooling System (CSCS) pond (i.e., the Ultimate Heat Sink) from 100 °F to 101.25 °F.

Date of issuance: August 2, 2007.

Effective date: August 2, 2007.

Amendment Nos.: 183 and 170.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications and License.

Public comments requested as to proposed no significant hazards consideration (NSHC): No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated August 2, 2007.

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Russell Gibbs.

Dated at Rockville, Maryland, this 20th day of August 2007.

For The Nuclear Regulatory Commission.

John W. Lubinski,

Acting Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. E7–16766 Filed 8–27–07; 8:45 am]

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