4. How can the burden of the information collection be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the draft supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O–1 F21, Rockville, MD 20852. OMB clearance requests are available at the NRC worldwide Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html.* The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Comments and questions about the information collection requirements may be directed to the NRC Clearance Officer, Brenda Jo. Shelton, U.S. Nuclear Regulatory Commission, T–5 F53, Washington, DC 20555–0001, by telephone at 301–415–7233, or by Internet electronic mail to *INFOCOLLECTS@NRC.GOV.*

Dated at Rockville, Maryland, this 15th day of June, 2005.

For the Nuclear Regulatory Commission.

Brenda Jo. Shelton,

NRC Clearance Officer, Office of Information Services.

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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 May 26, 2005, to June 9, 2005. The last biweekly notice was published on June 7, 2005 (70 FR 33210).

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, Rules and Directives 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 bases 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.*

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: May 25, 2005.

Description of amendment request: The amendment would revise Technical Specification Section 3.4.9, ''Pressurizer,'' to revise the pressurizer water level limit during operation in Mode 3 (hot standby).

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.

Pressurizer water level is an assumed initial condition for certain accident analyses. Plant initial conditions are not accident initiators and do not have an effect on the probability of the accident occurring. The proposed change only revises the specified limit on water level in the pressurizer, so this change does not affect accident probability.

Pressurizer water level is an assumed initial condition for accidents such as LOCA [loss-of-coolant accident], loss-of-load and loss-of-normal feedwater. The limiting accident analysis results occur at full power conditions when the available core thermal power is maximized. The proposed change does not affect the specified pressurizer level limit at any power level from zero to full power. That is, the pressurizer level limit is not being changed in Modes 1 and 2. The proposed change does revise the specified pressurizer water level limit in Mode 3 (Hot Standby) but this does not affect accident analysis results because the limiting analyses will remain those that are postulated to occur in Mode 1 with the plant at full power.

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 does not involve physical changes to existing plant equipment or the installation of any new equipment. The design of the pressurizer, the pressurizer level control system and the pressurizer safety valves is not being changed and the ability of these systems, structures, and components to perform their design or safety functions is not being affected. The proposed change revises the specified limit on pressurizer water level in Mode 3 (Hot Standby) to allow operators greater flexibility in performing a plant cooldown. The method used in performing the plant cooldown is not being changed. This proposed change does not create new failure modes or malfunctions of plant equipment nor is there a new credible failure mechanism.

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.

Pressurizer level is an initial condition assumed in certain accident analyses involving an insurge in the pressurizer and an increasing reactor coolant system (RCS) pressure. These analyses demonstrate that the design pressure for the RCS is not exceeded for the limiting analyses based on the plant at full power. The proposed change does not affect the existing Technical Specification requirement for Mode 1 (Power Operation) or Mode 2 (Plant Startup) and therefore does not affect the assumptions or results of these accident analyses. The margin for RCS design pressure demonstrated by these analysis results is not being reduced. The proposed change only applies to the pressurizer level limit in Mode 3 (Hot Standby) when there is substantially lower thermal energy available to cause rapid expansion of reactor coolant and an insurge to the pressurizer. Protection of the RCS pressure boundary is still maintained by the pressurizer safety valves, which are not being modified by the proposed change in pressurizer water level.

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: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Richard J. Laufer.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: March 15, 2005.

Description of amendment request: A change is proposed to revise the Waterford Steam Electric Station Unit 3 (Waterford 3) Technical Specification (TS) Section 4.4.4.4 to modify the steam generator tube inspection Acceptance Criteria for the ''Plugging or Repair Limit'' and the ''Tube Inspection,'' as contained in the Waterford 3 TS Surveillance Requirements (SR) 4.4.4.4.a.7 and 4.4.4.4.a.9, respectively. The purpose of these changes is to define the depth of the required tube inspections and to clarify the plugging criteria within the tubesheet region.

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.

Conducting the rotating Plus Point probe inspections to a minimum tubesheet length of 10.4 inches maintains the existing design limits and does not increase the probability or consequences of an accident involving tube burst or primary to secondary accident-induced leakage, as previously analyzed in the Waterford 3 Final Safety Analysis Report. Also the NEI [Nuclear Energy Institute] 97–06 structural integrity and accident induced leakage of the steam generator tubes performance criteria will continue to be satisfied.

Tube burst is precluded for a tube with defects within the tubesheet region because of the constraint provided by the tubesheet. As such, tube pullout resulting from the axial forces induced by primary to secondary differential pressures would be a prerequisite for tube burst to occur. Any degradation below C* is shown by empirical test results and analyses to be acceptable, thereby precluding an event with consequences similar to a postulated tube rupture event. WCAP–16208–P has shown that tube flaws below the C* length will not result in primary to secondary leakage greater than 0.1 gpm [gallons per minute] per steam generator. Inspection to the C* length will ensure that the postulated accident induced leakage for events that involve a faulted steam generator (*e.g.*, a main steam line break (MSLB)) will remain within both the current and proposed extended power uprate (EPU) accident analyses of 720 gpd (0.5 gpm) and 540 gpd (0.375 gpm), respectively.

Therefore, the proposed change does not affect the probability or consequences of any Waterford 3 analyzed accidents.

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

Response: No.

Steam generator tube leakage and structural integrity will be maintained during all plant conditions upon implementation of the proposed inspection scope and plugging or repair limit changes to the Waterford 3 Technical Specifications. These changes do not introduce any new mechanisms that might result in a different kind of accident from those previously evaluated. Even with the limiting circumstances of a complete circumferential separation (360o through wall crack) of all of the tubes below the C* length, tube pullout is precluded and leakage is predicted to be maintained within both the current and proposed extended power uprate (EPU) accident analyses assumptions.

Therefore, the proposed change does not create the possibility of a new or different type 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 inspection and plugging criteria will better assure that steam generator tube performance is maintained within its design basis and within the safety analysis assumptions. Operation with potential tube degradation below the C* inspection length within the tubesheet region of the steam generator tubing meets the intent of the inspection guidance of RG 1.83, *Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes*, the requirements of General Design Criteria 14, 30 and 32 of 10 CFR 50, and the recommendations of NEI–97–06, Steam Generator Program Guidelines. The total leakage from an undetected flaw population below the C* inspection length under postulated accident conditions is accounted for to assure that the leakage criterion is met and bounded by both the current and the proposed EPU accident analyses assumptions. Adequate margin remains for other possible steam generator tube leak sources.

The proposed changes also maintain the structural and accident-induced leakage integrity of the steam generator tubes as required by NEI 97–06 and the plant design basis.

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: N. S. Reynolds, Esquire, Winston & Strawn 1400 L Street NW., Washington, DC 20005–3502.

NRC Section Chief: David Terao.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1 (BVPS–1), Beaver County, Pennsylvania

Date of amendment request: April 11, 2005.

Description of amendment request: The proposed amendment would revise the BVPS–1 Technical Specifications (TSs) to permit operation with replacement Model 54F steam generators (SGs) installed. These include changes to reactor core safety limits, reactor trip system and engineered safety features actuation system setpoints, and other safety analysis inputs related to the proposed new model 54F steam generators as well as changes to steam generator limiting conditions for operation and surveillance requirements. These proposed TS changes were originally submitted as part of the licensee's extended power uprate application, dated October 4, 2004, however, delays in the review of that application have required the licensee to separately request these proposed TS changes in order to support SG replacement during and startup from the BVPS–1 2006 refueling outage.

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 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The safety and radiological dose consequence analyses confirmed that safety analysis and dose consequence analysis acceptance criteria will be satisfied for the Model 54F BVPS Unit No. 1 replacement steam generators, including changes to reactor core safety limits, reactor trip system (RTS) and engineered safety features actuation system (ESFAS) setpoints, and other safety analysis inputs related to the proposed changes. The analyses are conservative and bounding with respect to operation with RSGs [replacement steam generators] at the current licensed maximum power level.

For the purpose of this evaluation, the proposed changes to Technical Specifications 3.4.1.3, Reactor Coolant system Shutdown, and 3.4.5, Steam Generators, which will directly address the new Unit No. 1 replacement steam generators (RSG) can be grouped in the following areas:

(a) The first area of change is to remove the references to repair of tubes by sleeving since they are not applicable to the RSG tubes.

The accidents of interest are [steam generator] tube rupture and steam line break. A reduction in tube integrity could increase the possibility of a tube rupture accident and could increase the consequences of a steam line break. The tubing in the RSGs is designed and evaluated consistent with the margins of safety specified in the ASME Code [American Society of Mechanical Engineers, Boiler and Pressure Vessel Code], Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action if tube degradation is present. The basis for the 40% through wall plugging limit is applicable to the RSGs just as it was to the original steam generators (OSG). An analysis has been performed consistent with the guidance in Draft Regulatory Guide 1.121 to justify the applicability of the 40% through wall plugging limit. As a result, there is no reduction in tube integrity for the RSGs.

Elimination of the repair option and the associated references to repair of the OSG tubes is an administrative adjustment since the sleeve design is not applicable to the RSGs. The elimination of the repair option does not alter the requirements for inservice inspection or reduce the plugging limit for the RSG tubes.

(b) The second area of change is to remove the references to voltage-based repair criteria on tube-to-tube support plate intersections since they are not applicable to the RSG tubes.

Elimination of the repair option and the associated repair of the OSG tubes is an administrative adjustment since the voltage based repair criteria is not applicable to the RSGs. The elimination of the repair option does not alter the requirements for inservice inspection or reduce the plugging limit for the RSG tubes.

(c) The third area of change is to update the wording and content of the TS to provide clarification and to incorporate wording enhancements consistent with the updates made to the subject TS for several other plants that have replaced steam generators. Since the RSGs will be subjected to a preservice inspection prior to installation, there is no need to perform inservice inspection following installation.

The changes to update the wording and content of the TS to provide clarification and to incorporate wording enhancements are administrative changes that provide clarifications. These changes do not alter the requirements for inservice inspection or the plugging limit for the tubes.

(d) The fourth area of change is to revise the steam generator water levels.

The proposed steam generator water level setpoint changes do not impact the initiation of accidents; therefore, they do not involve an increase in the probability of an accident previously evaluated. The proposed changes do impact the safety analyses for accidents that credit the applicable trips and associated system actions; however, they do not alter these accidents or the associated accident acceptance criteria. The safety analyses for these accidents have been performed at 2900 MWt [megawatts thermal] (which is conservative and bounding for the current licensed power level of 2689 MWt) and show acceptable results. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

The proposed change to steam generator water level used to verify steam generator operability in Modes 4 and 5, *i.e.*, TS 3.4.1.3, does not impact the initiation of accidents; therefore, it does not involve an increase in the probability of an accident previously evaluated. The proposed change does not alter the safety analyses for accidents or the associated accident acceptance criteria. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed changes, due to the replacement steam generators, do not alter the requirements for tube inspection, tube integrity, or tube plugging limit, therefore they do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Use of the VIPRE computer code and the WRB–2M correlation at BVPS for departure from nucleate boiling (DNB) analysis for those Updated Final Safety Analysis Report (UFSAR) transients and accidents for which DNB might be a concern will not involve a significant increase in the probability or consequences of an accident previously evaluated for the following reasons. The code and correlation are evaluation tools that are independent of the probability of an accident. Use of the code and correlation establish DNB limits such that core damage will not occur during postulated design basis accidents. Thus, use of the code and correlation will not involve a significant increase in the consequences of an accident previously evaluated.

Use of the 1979 ANS [American Nuclear Society] Decay Heat + 2σ 4 model for MSLB [main steam line break] outside containment M&E [mass and energy] releases will not have a significant increase in the probability or consequences of an accident previously evaluated because the model is not an accident initiator.

The remaining changes, which include the changes to the Overtemperature ΔT and Overpower ΔT equations, the change to the charging pump discharge pressure, and the additions of WCAP–14565–P–A and WCAP–15025–P–A to the list of NRC approved methodologies in TS 6.9.5, will not involve a significant increase in the probability or consequences of an accident previously evaluated because none of the changes are accident initiators.

The RSG radiological analysis reflects an expansion of the selective application of the AST methodology and incorporation of the ARCHON96 methodology for on-site atmospheric dispersion factors. The radiological analysis concludes that normal operation of the BVPS Unit No. 1 with the RSGs with an atmospheric containment will not impact the unit's compliance with the normal operation operator exposure limits set forth in 10 CFR 20 [Title 10 of the Code of Federal Regulations, Part 20], or the public exposure limits set forth in 10 CFR 20, 10 CFR 50, Appendix I and 40 CFR 190, or with the post-accident exposure limits set forth by 10 CFR 100 or 10 CFR 50.67, as supplemented by Regulatory Guide 1.183, for the plant operator and the public.

The effects on accident radiation dose considered the replacement of the Unit No. 1 steam generators, a core power level to 2900 MWt, incorporation of the ARCHON96 methodology and the expansion of the selective implementation of the AST methodology. None of these changes are initiators of any design basis accident or event, and therefore, will not increase the probability of any accident previously evaluated. The probability of any evaluated accident or event is independent of these changes.

These proposed changes required alteration of some assumptions previously made in the radiological consequence evaluations. The assumption alterations were necessary to reflect the replacement steam generators for Unit No. 1 and the incorporation of the ARCHON96 and AST methodologies. These changes were evaluated for their effect on accident dose consequences. The updated dose consequence analyses demonstrate compliance with the limits set forth for AST applications in 10 CFR 50.67, as supplemented by Regulatory Guide 1.183 or 10 CFR part 100.

Therefore, in conclusion, none of the proposed changes involve a significant increase in the probability of an accident previously evaluated, and the dose consequences remain within the allowable limits set forth for AST applications in 10 CFR 50.67, as supplemented by Regulatory Guide 1.183 or 10 CFR part 100.

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 will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The areas of change described previously for the Unit No. 1 RSGs do not adversely affect the design or function of any other safety-related component. With respect to postulated accident conditions, the OSGs and the RSGs are the same. There is no mechanism to create a new or different kind of accident for the RSGs by eliminating repair criteria or by clarifying the applicability of inservice inspection requirements because a baseline of tube conditions is established and plugging limits are maintained to ensure that defective tubes are identified and removed from service.

The proposed changes to steam generator water level setpoints, and the steam generator water level used to verify steam generator operability in Modes 4 and 5 do not impact the initiation of accidents. They do not alter the accidents that credit the associated trips or accident acceptance criteria. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not alter the requirements for tube inspection, tube integrity, or tube plugging limit; therefore, they do not create the possibility of a new or different kind of accident from any previously evaluated.

Use of the VIPRE computer code and WRB–2M correlation at BVPS will not create the possibility of a new or different kind of accident from any accident previously evaluated because the code and correlation are evaluation tools. They are not accident initiators. Thus, their use cannot create a new or different kind of accident.

Use of the 1979 ANS Decay Heat $+ 2\sigma$ model for MSLB outside containment M&E releases will not create the possibility of a new or different kind of accident from any accident previously evaluated because the model does not alter how any equipment is operated.

The remaining changes, which include the changes to the Overtemperature ΔT and Overpower ΔT equations, the change to the charging pump discharge pressure, and the additions of WCAP–14565–P–A and WCAP–15025–P–A to the list of NRC approved methodologies in TS 6.9.5, will not create the possibility of a new or different kind of accident from any accident previously evaluated because these changes do not alter how any equipment is operated.

The radiological changes will not create the possibility of a new or different kind of accident from any previously evaluated because they do not affect how components or systems are operated, nor do they create new components or systems failure modes.

Therefore, in conclusion, none of the proposed changes 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 will not involve a significant reduction in a margin of safety.

The steam generator tube integrity provides the margin of safety. The tubing in the RSGs is designed and evaluated consistent with the margins of safety specified in the ASME Code, Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action if tube degradation is present. The basis for the 40% through wall plugging limit is applicable to the RSGs just as it was to the OSGs. A Regulatory Guide 1.121 analysis was performed to confirm the applicability of the 40% through wall plugging limit. As a result, there is no reduction in tube integrity for the RSGs.

The proposed changes to steam generator water level setpoints do not alter the reactor trip system/engineered safety features actuation system setpoint analysis methodology, or the associated accident analysis methodology or acceptance criteria. The safety analyses for these accidents have been performed at a power level of 2900 MWt (which is conservative and bounding for the current licensed power level of 2689 MWt) and show acceptable results. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The proposed change to the steam generator water level used to verify steam generator operability in Modes 4 and 5 does not alter the steam generator water level uncertainty and setpoint analysis methodology or the associated natural circulation analysis methodology or acceptance criteria. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The proposed changes to update the wording and content of the TS to provide clarification and to incorporate wording enhancements are administrative changes that provide clarifications.

The proposed changes do not alter the requirements for tube integrity, tube inspection or tube plugging limit; therefore, they do not involve a significant reduction in a margin of safety.

Use of the VIPRE computer code and the WRB–2M correlation at BVPS will not involve a significant reduction in a margin of safety because the code and correlation are used to establish a margin of safety previously approved by the NRC such that core damage will not occur.

Use of the 1979 ANS Decay Heat $+ 2\sigma$ model for MSLB outside containment M&E releases will not involve a significant reduction in a margin of safety because the results of the subject accident have been shown to produce acceptable results.

The remaining changes, which include changes to the Overtemperature σT and Overpower σT equations, the change to the charging pump discharge pressure, and the additions of WCAP–14565–P–A and WCAP–15025–P–A to the list of NRC approved methodologies in TS 6.9.5, will not involve a significant reduction in a margin of safety because they are being made to maintain the existing margin of safety.

The radiological changes will not involve a significant reduction in a margin of safety because BVPS compliance with the limits set forth in 10 CFR 20, 10 CFR 50, Appendix I, 40 CFR 190, 10 CFR 100 and 10 CFR 50.67, as supplemented by Regulatory Guide 1.183, will be maintained following approval of the requested changes.

A FENOC assessment of the cumulative effect of the proposed changes provides [a] reasonable expectation that collectively they will not result in a significant reduction in the overall margin of safety. The results of the analyses demonstrate that the applicable design and safety criteria and regulatory requirements will continue to be met following approval of the proposed changes.

Therefore, in conclusion, none of the propose changes 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: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard J. Laufer.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: June 1, 2005.

Description of amendment request: The amendments proposed by Southern Nuclear Operating Company (SNC) would revise the Technical Specifications (TS) to replace the previous TS requirement to implement a Containment Tendon Surveillance Program based on Regulatory Guide 1.35, Rev. 2, ''Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures,'' with a Containment Inspection Program that complies with the current requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a, ''Codes and Standards,'' in order to reflect the latest requirements for tendon surveillance.

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

The proposed change replaces the previous TS requirement to implement a Containment Tendon Surveillance Program based on Regulatory Guide 1.35, Rev. 2, with a Containment Inspection Program that complies with the current requirements of 10 CFR 50.55a. This regulation requires licensees to implement a Containment Inspection Program in compliance with the 1992 Edition with the 1992 Addenda of Subsection IWE, ''Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants,'' and with Subsection IWL, ''Requirements for Class CC Concrete Components of Light-Water Cooled Plants,'' of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) with additional modifications and limitations as stated in 10 CFR 50.55a(b)(2)(ix). SNC has implemented a Containment Inspection Program that complies with the regulatory requirements. This proposed TS amendment is requested to update the TS to the latest 10 CFR 50.55a regulatory requirements.

In addition, reporting requirements that are redundant to existing regulations are deleted, minor editorial changes are made, and the applicability of [Surveillance Requirement] SR 3.0.2 to the tendon surveillance program is deleted since surveillance frequencies and associated extensions are specified in ASME Section XI, Subsection IWL.

By complying with the regulatory requirements described in 10 CFR 50.55a, the probability of a loss of containment structural integrity is maintained as low as reasonably achievable. Maintaining containment structural integrity as described in the revised Containment Inspection Program does not impact the operation of the reactor coolant system (RCS), containment spray (CS) system, or emergency core cooling system (ECCS). The Containment Inspection Program ensures that the containment will function as designed to provide an acceptable barrier to release of radioactive materials to the environment. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not impact any accident initiators or analyzed events, nor does it impact the types or amounts of radioactive effluent that may be released offsite. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Maintaining containment structural integrity does not impact the operation of the RCS, CS system, or ECCS. The proposed change does not involve a modification to the physical configuration of the plant or a change in the methods governing normal plant operation. The proposed change does not introduce a new accident initiator, accident precursor, or malfunction mechanism. 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 license amendment does not involve a significant reduction in a margin of safety.

By complying with the regulatory requirements described in 10 CFR 50.55a, the probability of a loss of containment structural integrity is maintained as low as reasonably achievable. The Containment Inspection Program ensures that the containment will function as designed to provide an acceptable barrier to release of radioactive materials to the environment. The proposed change does not adversely affect plant operation or existing safety analyses. 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: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: Evangelos C. Marinos.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application request: May 26, 2005.

Description of amendment request: The amendment would change Technical Specification (TS) 3.7.2, ''Main Steam Isolation Valves (MSIVs),'' by adding the MSIV actuator trains to (1) the limiting condition for operation (LCO) and (2) the conditions, required actions, and completion times for the LCO. The existing conditions and required actions in TS 3.7.2 are renumbered to account for the new conditions and required actions.

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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to incorporate requirements for the MSIV actuator trains do not involve any design or physical changes to the facility, including the MSIVs and actuator trains themselves. The design and functional performance requirements, operational characteristics, and reliability of the MSIVs and actuator trains are thus unchanged. There is therefore no impact on the design safety function of the MSIVs to close (as an accident mitigator), nor is there any change with respect to inadvertent closure of an MSIV (as a potential transient initiator). Since no failure mode or initiating condition that could cause an accident (including any plant transient) evaluated per the FSAR [Callaway Final Safety Analysis Report]-described safety analyses is created or affected, the [proposed] change[s] cannot involve a significant increase in the probability of an accident previously evaluated.

With regard to the consequences of an accident and the equipment required for mitigation of the accident, the proposed changes involve no design or physical changes to the MSIVs or any other equipment required for accident mitigation. With respect to [the] MSIV actuator train allowed outage times [(*i.e.*, completion times)], the consequences of an accident are independent of equipment allowed outage times as long [as] adequate equipment availability is maintained. The proposed MSIV actuator train allowed outage times take into account the redundancy of the MSIV actuator trains and are limited in extent consistent with other allowed outage times specified in the Technical Specifications. Adequate equipment (MSIV) availability would therefore continue to be required by the Technical Specifications. On this basis, the consequences of applicable, analyzed accidents (such as a main steam line break) are not significantly impacted by the proposed changes. Based on all of the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated.

Response: No.

None of the proposed changes, *i.e.*, the addition of Conditions, Required Actions and Completion Times [and addition to the LCO] to [the] Technical Specifications for the MSIV actuator trains, involve a change in the design, configuration, or operational characteristics of the plant. No physical alteration of the plant is involved, as no new or different type of equipment is to be installed. The proposed changes do not alter any assumptions made in the safety analyses, nor do they involve any changes to plant procedures for ensuring that the plant is operated within analyzed limits. As such, no new failure modes or mechanisms that could cause a new or different kind of accident from any previously evaluated are being introduced.

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

3. [Do] the proposed change[s] involve a significant reduction in a margin of safety.

Response: No.

The proposed addition of Conditions, Required Actions and Completion Times [and proposed addition to the LCO] to the Technical Specifications for the MSIV actuator trains does not alter the manner in which safety limits or limiting safety system settings are determined. [There are no proposed changes to safety limits or limiting safety system settings.] No changes to instrument/system actuation setpoints are involved. The safety analysis acceptance criteria are not impacted by [these proposed] change[s], and the proposed change[s] will not permit plant operation in a configuration outside the design basis.

Therefore, the proposed changes do 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: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Robert A. Gramm.

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, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: October 21, 2004.

Description of amendment request: The amendment deletes the Technical Specification (TS) requirements to submit monthly operating reports and annual occupational radiation exposure reports. The change is consistent with Revision 1 of the Nuclear Regulatory Commission approved Technical Specifications Task Force (TSTF) Change Traveler, TSTF–369, ''Elimination of Requirements for Monthly Operating Reports and Occupational Radiation Exposure Reports.'' This TS improvement was published in the **Federal Register** (69 FR 35067) on June 23, 2004, as part of the Consolidated Line Item Improvement Process.

Date of issuance: June 8, 2005.

Effective date: June 8, 2005.

Amendment No.: 254.

Facility Operating License No. DPR–16: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. April 8, 2005 (70 FR 18056). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. Comments received from the State of New Jersey are discussed in Section 7.0 of the related safety evaluation. The notice also provided an opportunity to request a hearing by June 7, 2005, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 8, 2005.

Attorney for licensee: Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Richard J. Laufer.

Dominion Nuclear Connecticut, Inc., et al., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: September 7, 2004.

Brief description of amendment: The amendment revised the required frequency of quench and recirculation spray nozzle surveillances from once every 10 years to ''following maintenance which could result in nozzle blockage.'' The change also revised wording to correct grammar.

Date of issuance: May 31, 2005.

Effective date: As of the date of issuance and shall be implemented within 180 days from the date of issuance.

Amendment No.: 222.

Facility Operating License No. NPF–49: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 7, 2004 (69 FR 70715).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 2005.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: June 3, 2003, as supplemented by letter dated January 18 and May 10, 2005.

Brief description of amendments: The amendments would add a note to Limiting Condition of Operation 3.7.11, ''Auxiliary Building Filtered Ventilation Exhaust System (ABFVES),'' that would allow the Auxiliary Building pressure boundary to be opened intermittently under administrative control.

Date of issuance: June 2, 2005.

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

Amendment Nos.: 229 and 211.

Renewed Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 16, 2004 (69 FR 12365). The supplements dated January 18 and May 10, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 2, 2005.

No significant hazards consideration comments received: No.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: December 20, 2004.

Brief description of amendment: The amendment deletes the requirements related to monthly operating reports and occupational radiation exposure reports.

Date of issuance: May 25, 2005.

Effective date: As of the date of issuance, and shall be implemented 90 days from the date of issuance.

Amendment No.: 145.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 1, 2005 (70 FR 9990).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 25, 2005.

No significant hazards consideration comments received: No.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: December 7, 2004.

Brief description of amendment: This amendment revised the Technical Specifications (TSs) by removing the surveillance requirement (SR) for testing the setting of the standby liquid control system pressure relief valves. Also, the SR for the recirculation pump discharge valves was revised to remove stroke time specifications.

Date of Issuance: June 1, 2005.

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

Amendment No.: 224.

Facility Operating License No. DPR–28: The amendment revised the TSs.

*Date of initial notice in **Federal Register:*** January 18, 2005 (70 FR 2889).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated June 1, 2005.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: April 6, 2004, as supplemented by four letters dated April 15, 2005.

Brief description of amendments: The amendments convert the current Technical Specifications (CTS) to the improved Technical Specifications (ITS) and relocate license conditions to the ITS or other license controlled documents. The ITS are based on NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 30, 2001, and guidance provided in the Commission's Final Policy Statement, ''The U.S. Nuclear Regulatory Commission Final Policy Statement on Technical Specifications (TSs) Improvements for Nuclear Power Reactors,'' published on July 22, 1993 (58 FR 39132), and 10 CFR Part 50.36, ''TSs.'' The overall objective of the proposed amendments was to rewrite, reformat, and streamline the CTS to improve plant safety and the understanding of the bases underlying the TSs.

Date of issuance: June 1, 2005.

Effective date: As of the date of issuance and shall be implemented by October 30, 2005.

Amendment Nos.: 287, 269.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** September 29, 2004 (69 FR 58205). The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 1, 2005.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit No. 1, Oswego County, New York

Date of application for amendment: October 22, 2004.

Brief description of amendment: The amendment deleted Sections 5.3, ''Reactor Vessel,'' 5.4, ''Containment,'' and 5.6, ''Seismic Design,'' relocating all information, which pertains to design details, to the Updated Final Safety Analysis Report.

Date of issuance: June 6, 2005.

Effective date: As of the date of issuance to be implemented within 90 days.

Amendment No.: 189.

Facility Operating License No. DPR–63: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 7, 2004 (69 FR 70719).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 5, 2003, as supplemented by letter dated April 22, 2004.

Brief description of amendments: The amendments revised the Point Beach Nuclear Plant (PBNP), Units 1 and 2, Updated Final Safety Analysis Report [UFSAR] to reflect the Commission staff's approval of the WCAP–14439-P, Revision 2 analysis entitled, ''Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant Units 1 and 2 for the Power Uprate and License Renewal Program.''

Date of issuance: June 6, 2005.

Effective date: As of the date of issuance and shall be implemented with the next update of the UFSAR in accordance with 10 CFR 50.71(e).

Amendment Nos.: 219, 224.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the License.

Date of initial notice in ***Federal Register:*** February 7, 2005 (70 FR 6466). The supplement dated April 22, 2004, provided clarifying information that did not change the scope of the amendment, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 15, 2004.

Brief description of amendments: The amendments revised Technical Specifications related to the reactor coolant pump flywheel inspection program by increasing the inspection interval to 20 years.

Date of issuance: June 6, 2005.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 218, 223.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 29, 2005 (70 FR 15945).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 15, 2004.

Brief description of amendments: The amendments revise Technical Specifications related to the reactor coolant pump flywheel inspection program by increasing the inspection interval to 20 years.

Date of issuance: June 7, 2005.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment Nos.: 170, 160.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 15, 2005 (70 FR 12748).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 7, 2005.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: September 23, 2004, and its supplements dated December 21, 2004, and April 7, 2005.

Brief description of amendments: The amendments increase the current minimum emergency diesel generator fuel oil inventory required to be maintained onsite to support the use of low-sulfur fuel oil required by California Air Resources Board.

Date of issuance: May 25, 2005.

Effective date: As of the date of issuance, and shall be implemented within 90 days from the date of issuance.

Amendment Nos.: Unit 1—181; Unit 2—183.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 4, 2005 (70 FR 402). The December 21, 2004, and April 7, 2005, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 25, 2005.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: June 5, 2003, as supplemented by letters dated June 3 and October 26, 2004.

Brief description of amendments: The amendments authorize changes to the Updated Final Safety Analysis Report (UFSAR) for both units, to acknowledge credit for possible operator action to ensure that the containment design pressure is not exceeded in the event of a high energy line break inside containment with a consequential failure of the station control and service air system inside containment.

Date of issuance: May 24, 2005.

Effective date: As of the date of issuance and shall be implemented as part of the next UFSAR update made in accordance with 10 CFR 50.71(e).

Amendment Nos.: 302 and 292.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments authorize changes to the UFSAR.

*Date of initial notice in **Federal Register:*** June 24, 2003 (68 FR 37584). The supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 24, 2005.

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: October 27, 2004.

Brief description of amendment: The amendment revised Technical Specification 3.7.3, ''Main Feedwater Isolation Valves (MFIVs),'' to add the main feedwater regulating valves (MFRVs) and the associated MFRV bypass valves (MFRVBVs). In addition, the allowed outage time, or completion time, for inoperable MFIVs is extended.

Date of issuance: May 31, 2005.

Effective date: This amendment is effective as of its date of issuance, and shall be implemented prior to entry into Mode 3 in the restart from the upcoming Refueling Outage 14 (fall 2005).

Amendment No.: 167.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 7, 2004 (69 FR 70722).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 2005.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 10th day of June, 2005.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. E5–3138 Filed 6–20–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Report for Comment: ''Documentation and Applications of the Reactive Geochemical Transport Model RATEQ,'' NUREG/CR–6871

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability and request for comments.

Background

The U.S. Nuclear Regulatory Commission (NRC) uses environmental models to evaluate the potential release of radionuclides from NRC-licensed sites. In doing so, the NRC recognizes