initial scoping process, NRC issued a scoping report in September 1998.

NRC's initial scoping process was based on the description of the PFSF contained in the applicant's submittal of June 20, 1997, which did not include the proposed rail line on public land administered by BLM. This rail line proposal was submitted to NRC on August 28, 1998, as an amendment to the PFS application. Similarly, BIA's conditional approval of the proposed lease agreement was issued prior to the applicant's proposal of the rail line.

As a result of the applicant's August 28, 1998, revision of its transportation proposal, NRC, BIA, and BLM determined that additional scoping meetings should be conducted. Additional scoping meetings were held on April 29, 1999, in Salt Lake City, and Tooele City, Utah. The meetings were noticed in the **Federal Register** on April 14, 1999 (64 FR 18451). Primarily, the scoping meetings focused on environmental issues associated with the rail line proposed in the applicant's August 28, 1998, license application amendment, the request for issuance of a ROW over public lands managed by BLM, and environmental concerns associated with the proposed lease agreement that may not have been addressed in the NRC's initial scoping process. In addition, interested parties were also provided the opportunity to submit written comments. Following the additional scoping meetings and comment period, a supplemental scoping report was issued in November 1999.

Although STB was not identified as a cooperating agency during the scoping process, the environmental issues related to its federal action (*i.e.*, approving the construction and operation of the proposed rail line) were discussed during the scoping process. STB has determined that these scoping activities provided sufficient opportunity for the public to comment on the proposed action and the scope of the EIS. Interested parties will have an opportunity to provide comments on the draft EIS.

Dated at Rockville, Maryland, this 27th day of January 2000.

For the Nuclear Regulatory Commission.

E. William Brach,

Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 00–2930 Filed 2–8–00; 8:45 am]

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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 Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision 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 January 14, 2000, through January 28, 2000. The last biweekly notice was published on January 26, 2000 (65 FR 4268).

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.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing 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 Review and Directives Branch, Division of Freedom of Information and Publications 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 NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By March 10, 2000, 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 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the

Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, 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 factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of 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. Petitioner must provide 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 to relief. A petitioner who fails to file such a supplement which satisfies 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, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, 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 with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Dated of amendments request: January 25, 2000.

Description of amendments request: The proposed amendment requests a revision to the definition of Response Time Testing (RTT) for the Reactor Protective System (RPS) and Engineered Safety Features Actuation System (ESFAS). The revision allows use of either an allocated sensor response time or a measured sensor response time for pressure sensors used in channels of RPS and ESFAS. The request is based on Combustion Engineering NPSD–1167, Revision 1, ''Elimination of Pressure Sensor Response Time Testing Requirements—CEOG Task 1070.''

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 licensing basis change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the safety analysis report.

This change to the licensing basis does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same Reactor Protective System and Engineered Safety Features Actuation System instrumentation is being used; the time response allocations/modeling assumptions in Updated Final Safety Analysis Report Chapter 14 analyses remain the same; only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the Updated Final Safety Analysis Report. Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

2. The proposed licensing basis change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the safety analysis report.

This change does not alter the performance of the pressure and differential pressure sensors used in the plant protection systems. These sensors will still have their response time verified before they are placed in operational service and after any maintenance to them that could affect their response time. Changing the method of periodically verifying instrument response for certain sensor (assuring equipment operability) from time response testing to calibration, use of actual data, and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the sensor response characteristic. Implementation of the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed licensing basis change does not involve a significant reduction in margin of safety.

The total Reactor Protective System and Engineered Safety Features Actuation System response time assumed in the safety analysis is not affected by this change. The periodic system response time verification method for selected pressure and differential pressure sensors is modified to allow the use of allocated data based on actual test results or other verifiable response time data. Verification methods and calibration tests assure that any degradation sufficient to significantly affect sensor response time will be detected before the total system response time exceeds that defined in the safety analysis. Therefore, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant 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 amendments request involves no significant hazards consideration.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Marsha Gamberoni, Acting.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Dated of amendment request: November 18, 1999.

Description of amendment request: The proposed amendment would remove license condition 3.H, ''Long Term Program,'' from Facility Operating License DPR–35 for the Pilgrim Nuclear Power Station.

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. No physical changes to the facility will occur as a result of this amendment. Work activities will continue to receive the appropriate level of review in accordance with Pilgrim procedures and practices. The organizational structure and processes that control and manage these activities ensure activities are prioritized and performed in a manner consistent with plant safety. The proposed amendment removes an administrative burden that is no longer required.

(2) The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. No changes to the physical design and operation of the plant will occur as a result of this amendment. The processes by which activities are planned, prioritized, and controlled are not affected. The appropriate level of technical review and management oversight will continue to be performed in accordance with existing procedures and practices to ensure activities are performed in a manner consistent with plant safety.

(3) The proposed amendment does not involve a significant reduction in a margin of safety. As stated earlier, no changes to the physical design and/or operation of any plant systems will occur as a result of this amendment; therefore, there is no reduction in any margins of safety. Work activities will continue to receive the appropriate technical review and management oversight to ensure activities are prioritized and performed in a manner consistent with plant safety. The proposed amendment removes an administrative burden that is no longer required.

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

Attorney for licensee: W. S. Stowe, Esquire, Entergy Nuclear Generation Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Section Chief: James W. Clifford.

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

Dated of amendment request: November 29, 1999

Description of amendment request: The proposed amendment would relocate the requirements associated with the high-steam-generator-level trip functions of the Reactor Protective System from the Technical Specifications to the Technical Requirements Manual.

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—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The Steam Generator Level—High function of the RPS [Reactor Protection System] is not credited in any accident analyses nor does it correspond to any TS [Technical Specification] Safety Limit. The high-level function acts to protect the Main Turbine from excessive moisture carryover during feedwater transient events. Protection of the Main Turbine is not required to adequately assure continued reactor safety or the health and safety of the public. Although this function may also serve to limit water intrusion into the main steam lines and consequential overcooling events, its role in this capacity is insignificant, as it does not directly act to secure feedwater from the steam generators. This Steam Generator Level—High function acts only to isolate the Main Turbine from the steam generators by causing a reactor trip, which in turn actuates a turbine trip. This function does not meet any of the criterions listed in 10 CFR 50.36(c)(2) (ii) for inclusion into the technical specifications for ANO–2 [Arkansas Nuclear One, Unit 2], and, therefore, may be excluded from the TSs. Since no changes are made that affect the current operation of this function during its relocation to the ANO–2 TRM [Technical Requirements Manual], and because this function is not credited in any accident analyses, no increase in the probability or consequences of an accident previously evaluated is evident.

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

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

The proposed changes relocate affected TS requirements associated with the Steam Generator Level—High Functions of the RPS from the ANO–2 TSs to the ANO–2 TRM. Future revisions to the setpoints and values associated with this function will be established within the requirements of 10 CFR 50.59 to ensure that excessive moisture carryover is prevented in order to protect the turbine and steam line loads. The Steam Generator Level—High Trip setpoint is not credited in any accident analyses and performs only an equipment protection function. The setpoint continues to protect the Main Turbine from damage and preserves operating margin to accommodate excessive feedwater flow prior to trip.

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

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

The Steam Generator Level—High Trip setpoint is not credited in any accident analyses and performs only an equipment protection function. The setpoint continues to protect the Main Turbine from damage and preserves operating margin to accommodate excessive feedwater flow prior to trip. In addition, turbine failure has been previously evaluated at ANO–2 as not to be a significant threat to the health and safety of the public.

Events that may result from water intrusion into the main steam lines have been previously evaluated and found not to rely upon the Steam Generator Level—High Trip function. The relocation of the requirements associated with the Steam Generator Level—High function from the TSs to the ANO–2 TRM does not change the current values and requirements. Since no technical change in the setpoint or allowable value is proposed by this submittal and because the Steam Generator Level—High function does not meet any of the four criterion of 10 CFR 50.36(c)(2)(ii), no significant change to the margin of safety is evident.

Therefore, this change 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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Dated of amendment request: November 29, 1999 .

Description of amendment request: The proposed amendment would revise selected Technical Specifications (TSs), Bases, and portions of the Safety Analysis Report (SAR) to maintain consistency with the transient and accident analyses which evaluated the impact of the replacement steam generators (SGs) that are being used for Cycle 15 operation. TS changes are proposed for the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) low pressurizer pressure setpoints, the RPS and ESFAS low SG pressure setpoints, the RPS and ESFAS low SG level setpoints, the reactor coolant flow rate limit, and the high linear power trip setpoints with inoperable main steam safety valves (MSSVs). SAR changes would support the new TS values and would also include small increases in calculated offsite radiological doses using newer, more conservative methods, for some non-loss-of-coolant accident events. The doses would remain within the 10 CFR Part 100 acceptance criteria. The proposed amendment would also make changes to the TSs and Bases that are not directly related to the replacement SGs. These changes would revise the allowed outage time of the MSSVs in Modes 1 and 2 to allow up to 12 hours to reduce the high linear power level-high trip setpoint when one or more MSSVs are inoperable, and would revise the action statement in Mode 3 to maintain at least two MSSVs operable on each SG.

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—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed changes to the ANO–2 [Arkansas Nuclear One, Unit 2] TSs are analytically based which change setpoints and procedure limits. No physical modifications are required as a result of the proposed changes. The RPS/ESFAS setpoint changes provide functionally equivalent protection with the RSGs [replacement steam generators] as the previous setpoint values provided with the OSGs [original steam generators]. Proposed changes in regard to RCS [reactor coolant system] flow rate and High Linear Power Trip setpoints associated with conditions where MSSVs are inoperable represent appropriate restrictions that have resulted from the various analyses performed in support of RSG installation. An Emergency Core Cooling System (ECCS) performance analysis was performed to demonstrate conformance to 10 CFR 50.46 for operation with RSGs. For the large break Loss of Coolant Accident (LOCA), the most limiting single failure of the ECCS [is no failure to the ECCS]. The small break LOCA analysis was reanalyzed using the existing Supplement 2 Model (S2M) of the ABB CENP [ABB Combustion Engineering Nuclear Power] small break LOCA evaluation model. The analysis was performed for 0.03 ft^2, 0.04 ft^2, and 0.05 ft^2 in the reactor coolant pump (RCP) discharge leg. The results of both analyses demonstrate continued conformance to the ECCS acceptance criteria of 10 CFR 50.46. Non-LOCA analyses intended to confirm the Chapter 15 events in the ANO–2 SAR were also performed. The analyses were performed considering the proposed Safety Limits and the Limiting Safety Settings of the TSs and were confirmed to be bounding for the affected safety analyses. The results of the non-LOCA analyses indicate that operation with the RSGs in service is acceptable. As a result of the analyses and evaluations performed in support of the RSGs, the ANO specific safety parameters and regulatory limits are protected. Therefore, the proposed TS changes will not significantly increase the probability of an accident previously analyzed.

Loss of Coolant Accidents (LOCAs) and non-LOCA safety analyses supporting the proposed changes have been performed and have demonstrated conformance with all applicable Licensing Basis acceptance criteria. Although calculated radiological doses using newer, more conservative methods increase for some non-LOCA events (requiring a revision to Chapter 15 of the SAR), the results are within the acceptance criteria of 10 CFR 100. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

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

The proposed changes to the ANO–2 TSs are analytically based and require changing plant setpoints and procedural limits. No physical modifications are required as a result of the proposed changes. The RPS/ESFAS setpoint changes provide functionally equivalent protection with the RSGs as the previous setpoint values provided with the OSGs. Proposed changes in regard to RCS flow rate and High Linear Power Trip setpoints associated with conditions where MSSVs are inoperable represent appropriate restrictions that have resulted from the various analyses performed in support of RSG installation. The additional 8 hours provided for reducing the High Linear Power Level trip setpoints is acceptable due to the low probability of an event occurring within this period, based on operating experience which indicates such a time period is reasonable to complete the changes, and to provide consistency with the RSTS [Revised Standard Technical Specifications]. Therefore, the proposed TS changes will not create the possibility of a new or different kind of accident than previously analyzed.

A review of both LOCA and non-LOCA events was performed which confirms that existing licensing basis methodologies have been considered and that a new accident event has not been created.

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

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

LOCA and non-LOCA safety analyses supporting the proposed changes have been performed and have demonstrated conformance within applicable acceptance criteria. With the increased size of the RSGs and the change in design characteristics, the bases for the setpoints in the ANO–2 TSs are affected. However, based on the new analyses and evaluations conducted in support of this license amendment, the new TS setpoints provide adequate margin to protect established safety and regulatory limits. Although calculated offsite radiological doses increase slightly for some non-LOCA events documented in Chapter 15 of the ANO–2 SAR, the increases are not considered to be significant in that the results remain within the 10 CFR 100 acceptance criteria.

Therefore, this change 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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Dated of amendment request: July 29, 1999, as supplemented by letters dated August 8 and August 24, 1999 (NPF–38–220).

Description of amendment request: The proposed change modifies Technical Specifications (TS) 3.8.1.1 and associated Bases by extending the Emergency Diesel Generator allowed outage time from 72 hours to ten days. Additionally, this proposed change adds Section 6.16, ''Configuration Risk Management Program'' to the Administrative Controls of 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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: The emergency diesel generators (EDGs) are backup alternating current power sources designed to power essential safety systems in the event of a loss of offsite power. EDGs are not an accident initiator in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The EDGs provide backup power to components that mitigate the consequences of accidents. The proposed changes to allowed outage times (AOTs) do not affect any of the assumptions used in deterministic safety analyses.

In order to fully evaluate the EDG AOT extension, probabilistic safety analysis methods were utilized. The results of these analyses indicate no significant increase in the risk of an accident previously evaluated. These analyses are detailed in CE NPSD–996, Combustion Engineering Owners Group ''Joint Applications Report for Emergency Diesel Generators AOT Extension.''

The Configuration Risk Management Program is an Administrative Program that assesses risk based on plant status. Adding the requirement to implement this program for Technical Specification 3.8.1.1 ACTION b does not affect the probability or the consequences of an accident.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: The proposed change does not change the design or configuration of the plant. No new method of plant operation is involved.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: The proposed changes do not affect the Technical Specification limiting conditions for operation or their bases which support the deterministic analyses used to establish the margin of safety. Evaluations used to support the requested Technical Specification changes have been demonstrated to be either risk neutral or risk beneficial depending on precise plant conditions. These evaluations are detailed in CE NPSD–996.

Therefore, the proposed change will 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: Robert A. Gramm.

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

Dated of amendment request: July 29, 1999, as supplemented by letter dated August 24, 1999 (NPF–38–221).

Description of amendment request: The proposed change modifies Technical Specifications (TS) 3.6.2.1 to extend the allowable outage time to seven days for one Containment Spray System (CSS) train inoperable. A new ACTION has been added to provide a shutdown requirement for the inoperability of two CSSs. Additionally, the APPLICABILITY is being changed to provide an end state of MODE 4. Associated TS Bases changes are included.

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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: The Containment Spray System (CSS) is part of the Containment Depressurization and Cooling System. Inoperable CSS components are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

The CSS system is primarily designed to mitigate the consequences of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). These proposed changes do not affect any of the assumptions used in the deterministic LOCA or MSLB analyses. Hence the consequences of accidents previously evaluated do not change.

In order to fully evaluate the CSS AOT [Allowed Outage Time] extension, probabilistic safety assessment (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. These analyses are detailed in report CE NPSD–1045, ''Modifications To The Containment Spray System, and Low Pressure Safety Injection System Technical Specifications.''

The Configuration Risk Management Program is an Administrative Program that assesses risk based on plant status. Adding the requirement to implement this program for Technical Specification 3.6.2.1 does not affect the probability or the consequences of an accident.

Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. Allowing an extended AOT or changing the APPLICABILITY does not increase the probability that a failure leading to an analyzed event will occur. The CSS components are passive until an actuation signal is generated. This change does not increase the failure probability of the CSS components. As such, the probability of occurrence for a previously analyzed accident [is] not significantly increased.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: The proposed change does not change the design or configuration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, and the setpoints at which protective or mitigative actions are initiated are unaffected by this change. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed change will only provide the plant some flexibility in the AOT and chang[es] the APPLICABILITY. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: The proposed changes do not affect the limiting conditions for operation or their bases used in the deterministic analysis to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrate that the changes involve no significant increase in risk. These evaluations are detailed in report CE NPSD–1045. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. None of these are adversely impacted by the proposed change. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating a transient event. The proposed change, which allows operation to continue for up to 7 days with components inoperable in one CSS train, is acceptable based on the remaining CSS components providing 100% of the required CSS flow. The reduced potential for a self-induced plant transient resulting from unit shutdown required for a second inoperable CSS train is minimized. Therefore, the change does not involve a significant reduction in the margin of safety, and is offset by minimizing the potential for a self-induced plant transient.

Therefore, the proposed change will 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: Robert A. Gramm.

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

Date of amendment request: October 18, 1999.

Description of amendment request: The proposed change modifies Technical Specification (TS) 3.6.2.2 Limiting Condition for Operation to allow Waterford Steam Electric Station, Unit 3 to operate with two independent trains of containment cooling, consisting of one cooler per train, operable during modes 1, 2, 3, and 4. Associated changes to the TS Bases have been proposed.

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. Will the operation of the facility in accordance with these proposed changes involve a significant increase in the probability or consequence of an accident previously evaluated?

Response: The proposed change to Technical Specification (TS) 3.6.2.2 reduces the number of Containment Fan Coolers (CFC) from two to one required to be operable in each train of the Containment Cooling System for modes 1, 2, 3, and 4. This change does not create any new system interactions and has no impact on operation or function of any system or equipment in a way that could cause an accident. The CFCs are not an initiator of any events nor affect any accident initiators of any events analyzed in Chapter 15 of the UFSAR [Updated Final Safety Analysis Report]. Therefore this change will not impact the probability of occurrence of an accident.

The results of the reanalysis of the limiting Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) accidents show that the consequences of an accident previously evaluated are not increased by the change in the required number of operable CFCs. The limiting accidents affected by the proposed changes are identified below:

The peak containment pressure following the limiting LOCA (Double Ended Hot Leg Slot Break with minimum safety injection flow) was determined to be 35.2 psig [pounds per square inch, gauge] as compared to the current licensing basis limiting LOCA (Double Ended Suction Leg Slot Break with minimum safety injection flow) peak pressure of 43.1 psig.

The peak containment pressure at 24 hours following the start of the limiting LOCA (Double Ended Discharge Leg Slot Break with minimum safety injection flow) and the operation of one containment spray train and one partially flooded CFC operable was determined to be 15.5 psig with a peak pressure of 33.27 psig as compared to the current licensing basis of 14.9 psig with a peak pressure of 42.9 psig. The current licensing basis limiting LOCA is the Double Ended Suction Leg Slot Break with maximum safety injection flow and the operation of one containment spray train and two operable CFCs.

The peak containment pressure following the limiting MSLB (102% power with failure of one containment heat removal train consisting of one containment spray pump and one CFC operable) was determined to be 42.68 psig as compared to the current licensing basis peak pressure of 42.9 psig. The current licensing basis limiting MSLB is 75% power with the failure of one train of containment heat removal system consisting of one containment spray train and two operable CFCs.

The peak containment equipment qualification temperature following the limiting MSLB (102% power with the failure of one MSIV to close) was determined to be 397.4 °F as compared to the current licensing basis peak temperature of 409.1 °F. The current licensing basis limiting MSLB is 102% power with two CFCs per train operable and the failure of one train of containment spray.

These values above demonstrate that the containment design basis pressure and equipment qualification temperature of 44 psig and 413.5 °F, respectively, are not exceeded and the containment pressure at 24 hours after start of the limiting LOCA is less than 50% of the peak pressure.

The results of the containment response analysis discussed above satisfy the following NRC Staff Standard Review Plan (SRP) section 6.2.1.1.A guidance document acceptance criteria for a PWR [Pressurized Water Reactor] dry containment.

The peak calculated containment pressure following a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB) should be less than the containment design pressure.

To satisfy the requirements of GDC [General Design Criteria] 38 to rapidly reduce the containment pressure, the containment pressure should be reduced to less than 50% of the peak calculated pressure for the design basis LOCA within 24 hours after the postulated accident.

Thus, revising the containment cooling system TS to require only one operable CFC per train results in acceptable containment response and therefore, will not adversely impact the consequences of accidents previously evaluated.

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

2. Will the operation of the facility in accordance with these proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: The proposed change to reduce the number of operable Containment Fan Coolers (CFC) from two to one in each train of the Containment Cooling System for modes 1, 2, 3, and 4 does not alter the operation of the CFCs. Although only one of the two CFCs per train is required to be operable, the manner in which the CFCs perform their safety function is not changed. All four CFCs (two per train) will be maintained operable to the extent possible to provide the greatest defense in depth and operating flexibility.

This proposed change does not involve a change in plant design, nor does it involve any potential initiating events that would create any new or different kind of accident. This proposed change does not alter the way in which the plant is operated in a manner that would create a new or different accident. Therefore, since no hardware modifications will be made, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will the operation of the facility in accordance with these proposed changes involve a significant reduction in a margin of safety?

Response: The proposed change revises TS 3.6.2.2, Containment Cooling System. This change revises the required number of fan coolers from two fan coolers per train to one fan cooler per train. As described in the containment depressurization and cooling system Technical Specification Bases, the containment cooling system is designed to maintain the post accident containment peak pressure below its design value of 44 psig. The system is also designed to reduce the containment pressure by a factor of [two] from its post-accident peak within 24 hours.

The analyses that have been performed to support this Technical Specification change have shown that the peak containment pressure remains below 44 psig, the 24-hour containment pressure is less than half the peak pressure, and the containment peak temperature remains below the maximum temperature of 413.5 °F provided in the Bases for Technical Specifications 3.6.2.1 and 3.6.2.2. In comparison of the current safety margins to the safety margins that would exist if the proposed changes were in effect, the results of the analyses, illustrated below, show an increase in the margin of safety for containment pressure and equipment qualification temperature following the associated limiting LOCA and MSLB.

The peak containment pressure following the limiting LOCA was determined to be 35.2 psig as compared to the current licensing basis limiting LOCA peak pressure of 43.1 psig.

The peak containment pressure at 24 hours following the start of the limiting LOCA was determined to be 15.5 psig with a peak pressure of 33.27 psig as compared to the current licensing basis of 14.9 psig with a peak pressure of 42.9 psig.

The peak containment pressure following the limiting MSLB was determined to be 42.68 psig as compared to the current licensing basis peak pressure of 42.9 psig.

The peak containment equipment qualification temperature following the limiting MSLB was determined to be 397.4 °F as compared to the current licensing basis peak temperature of 409.1 °F.

This proposed change does not adversely impact a margin of safety, involve a change in plant design, or have any affect on the plant protective barriers. Therefore, the proposed change will not involve a significant reduction in the margin of safety.

The Nuclear Regulatory Commission 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: Robert A. Gramm.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: December 13, 1999.

Description of amendment request: The licensee proposes to change the license to delete an expired license condition and to make some editorial and administrative changes to correct or clarify the license.

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

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. In addition, the proposed changes do not affect the manner in which the plant responds in normal operation, transient or accident conditions nor do they change any of the procedures related to operations of the plant. The proposed changes do 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 acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative and editorial in nature and only correct, update and modify the Operation License.

The proposed changes do not affect the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Seabrook Station UFSAR. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures.

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

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

The proposed changes are administrative in nature and only correct, update and clarify the Seabrook Station Operating License. The proposed changes do not modify the facility nor do they modify the manner in which the plant will be operated nor do they affect the plant's response to normal, transient or accident conditions. The changes do not introduce a new mode of plant operation. The plant's design basis are not revised and the current safety analyses will remain in effect and the plant will continue to be operated in accordance with the existing Technical Specifications. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The proposed changes are administrative and editorial changes to the Seabrook Station Operating License that do not revise the Technical Specifications or the bases for the Technical Specifications. The safety margins established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits as specified in the Technical Specifications are not revised nor is the plant design or its method of operation revised by the proposed changes. Since there will be no changes to the physical design or operation of the plant, 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: March 18, 1998.

Description of amendment request: The proposed amendment would revise the Technical Specifications 2.15(4) and 2.15(5) to identify (1) all indication functions and control functions required for the alternate (remote) shutdown system (alternate shutdown panel and auxiliary feedwater panel), (2) panel locations of the functions, and (3) the number of channels required.

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

The proposed changes to Technical Specifications Section 2.15(4) and 2.15(5) identify functions, instruments, and controls along with their location and the number of required channels. The new Technical Specifications section addresses the regulatory requirements for equipment required for Alternative and Dedicated Shutdown Capability per 10 CFR Part 50 Appendix R. It will ensure that proper Limiting Conditions for Operation are entered for equipment or functional inoperability. There are no physical alterations being made to the Alternate Shutdown Panel and the Auxiliary Feedwater Panel or related systems. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not result in any physical alterations to the Alternate Shutdown Panel or the Auxiliary Feedwater Panel, or any plant configuration, systems, equipment, or operational characteristics. There will be no changes in operating modes, or safety limits, or instrument limits. With the proposed changes in place, Technical Specifications retain requirements for the Alternate Shutdown Panel and the Auxiliary Feedwater Panel. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes clarify the regulatory requirements for the Alternative and Dedicated Shutdown Capability as defined by 10 CFR Part 50, Appendix R. The proposed changes will not alter any physical or operational characteristics of the Alternate Shutdown Panel and the Auxiliary Feedwater Panel and their associated systems and equipment. Therefore, the proposed changes do not involve a 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: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: November 29, 1999.

Description of amendment request: The amendment would adopt the ''Standard Test Method for Nuclear Grade Activated Carbon'' for charcoal filter laboratory testing with certain exceptions.

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 license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The adoption of the new test method and acceptance criteria of ASTM [American Society for Testing and Materials] D3803–1989, with the exceptions as identified in the Technical Specifications, for activated charcoal filters does not involve any modifications to the plant, will not require changes to how the plant is operated nor will it affect the operation of the plant. Adoption of these provisions ensures compliance with the new test standard of ASTM D 3803–1989. Adoption of new test method will not cause an accident and therefore cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The adoption of the new test method and acceptance criteria of ASTM D 3803–1989, with the exceptions as identified in the Technical Specifications, for activated charcoal filters does not involve any modifications to the plant, will not require changes to how the plant is operated nor will it affect the operation of the plant. Adoption of new test method will not cause an accident and therefore cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed license amendment involve a significant reduction in a margin of safety?

Response: The proposed license amendment does not involve a significant reduction in a margin of safety. The use of outdated test protocols or inappropriate test conditions can lead to an overestimation of the charcoal filters' ability to adsorb radioiodine following an accident. The adoption of the new test method and testing criteria of ASTM D 3803–1989, with the exceptions as identified in the Technical Specifications, for activated charcoal filters would ensure at least a safety factor of two is maintained. Thus, the proposed change 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 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. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni, Acting.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: November 19, 1999.

Description of amendment request: The proposed amendment would revise Sections 4.7.A and 4.11.B of the Appendix A Technical Specifications (TSs) to the James A. FitzPatrick Operating License to adopt the surveillance test methods and performance criteria detailed in NRC Generic Letter 99–02 for laboratory testing of nuclear-grade charcoal. The proposed amendment also would reduce the minimum allowable Standby Gas Treatment System (SGTS) and Control Room Emergency Ventilation Air Ventilation Supply System (CREVASS) charcoal filter efficiencies specified in the TSs to those assumed in the updated radiological dose calculations.

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:

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92(c) since it would not:

(1) Involve an increase in the probability or consequences of an accident previously evaluated.

These changes are not modifications to the plant. They will not require changes to how the plant is operated, nor will they affect the operation of the plant.

Changes to these test methods will not cause an accident and therefore cannot increase the probability of an accident.

Calculated radiological doses increase as a result of reductions in assumed charcoal efficiencies, but remain within regulatory limits. Radiological doses at the site boundary and low population zone are less than 25 percent of 10 CFR 100 criteria. Post-accident doses to control room operators are also within regulatory limits.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

These changes are not modifications to the plant, nor will they require changes to how the plant is operated. Changes to these test methods will not cause an accident, and therefore cannot create the possibility of a new or different kind of accident.

(3) Involve a significant reduction in a margin of safety.

Evidence has been presented which contend that activated charcoal testing performed to test standards other than ASTM [American Society for Testing and Materials] D3803–1989 may be inaccurate and may overestimate its adsorption capabilities. The adoption of ASTM D3803–1989 charcoal performance standards and plant-specific test parameters ensures adequate safety margins.

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. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni, Acting.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: December 20, 1999.

Description of amendment request: This application for an amendment to the James A. FitzPatrick Technical Specifications (TS) proposes a change to the Main Steam Isolation Valve (MSIVs) closure scram setpoint. The proposed amendment changes the MSIV closure scram Trip Level Setting from ≤10% to ≤15% valve closure.

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 change will not significantly increase the probability or consequences of any previously evaluated accidents.

This proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. This proposed change to the MSIV scram trip setpoint from 90% open to 85% open results in the MSIV closure scram event having a slight delay in the initiation of the reactor scram. The MSIV scram event is not an accident initiator and will not increase the probability or consequence of any accident previously evaluated. An evaluation of events concluded that the MSIV direct scram remains a relatively low consequence event and has no effect on any operating limits. The limiting event analyzed for the reactor vessel overpressure event in the James A. FitzPatrick Updated Final Safety Analysis Report (UFSAR) is the MSIV closure terminated by a high neutron flux scram, which does not take credit for the MSIV closure valve position scram. In addition, an evaluation of the Main Steam Line Break outside containment event concludes that there is no impact on PCT [Peak Clad Temperature] or break flow and that the results presented in the UFSAR are bounding. An evaluation of the impact on containment was also made. The containment response is evaluated for much more severe events such as a Loss of Coolant Accident (LOCA) or stuck open Safety Relief Valve (SRV), thus the change in MSIV scram setpoint has no impact on the containment analysis. Therefore, changing the MSIV closure scram setpoint from 90% open to 85% open does not significantly increase the probability or consequences of any previously evaluated accidents.

2. The proposed change will not create the possibility of a new or different kind of accident.

This proposed change to the MSIV scram trip setpoint from 90% open to 85% open results in the MSIV closure scram event having a slight delay in the initiation of the reactor scram, and does not introduce a new or different kind of accident previously analyzed. An evaluation of the event determined that the MSIV closure scram remains a relatively low consequence event and has no effect on any operating limits. The limiting event analyzed for the reactor vessel overpressure event in the UFSAR is the MSIV closure terminated by a high neutron flux scram, which does not take credit for the MSIV closure valve position scram. The proposed change will not create the possibility of a new or different kind of accident.

3. The proposed change will not involve a significant reduction in a margin of safety.

This proposed change to the MSIV scram trip setpoint from 90% open to 85% open results in the MSIV closure scram event having a slight delay in the initiation of the reactor scram. An evaluation of the event determined that the MSIV closure scram remains a relatively low consequence event and has no effect on any operating limits. In addition, an evaluation of the Main Steam Line Break outside containment event concludes that there is no impact on PCT or break flow and that the results presented in the UFSAR are bounding. An evaluation of the impact on containment was also made. The containment response is evaluated for much more severe events such as a LOCA or stuck open SRV, thus the change in MSIV scram setpoint has no impact on the containment analysis. Changing the MSIV valve position scram setpoint from ≤10% to ≤15% of valve closure will allow the limit switches to be positioned such that both scram and indicating limit switches can be coordinated and provide accurate and reliable valve position indication in the control room. Therefore, changing the MSIV closure scram setpoint from 90% open to 85% open does not involve a significant reduction in a margin of safety and provides a net benefit to plant operations.

The proposed change will not increase the probability or consequences of any previously analyzed accident, introduce any new or different kind of accident previously evaluated, or significantly reduce existing margin to safety. Therefore, the proposed license amendment will 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: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni, Acting.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia.

Date of amendment request: August 30, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification 5.2.2 in order to raise the level of the approval authority for deviations from the guidelines provided to minimize unit staff overtime.

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?

The proposed license amendment would strengthen the administrative controls that permit plant personnel to work beyond those limits outlined in the TS's. As a result, there will be greater scrutiny on the amount of overtime being utilized to perform safety-related function. Therefore, it has been determined that operation of the facility in accordance with the proposed amendment will 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?

There will be no physical changes to the systems, components or structure of the facility as a result of this proposed license amendment. The initial assumptions of the design accident analyses will be unaffected. Therefore, operation of the facility 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. Do the proposed changes involve a significant reduction in a margin of safety?

This amendment raises the administrative level of management approval required for overtime in excess of the limits outlined in the TS. Therefore, operation of the facility in accordance with the proposed amendment will 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. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: October 4, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification 5.5.6, ''Prestressed Concrete Containment Tendon Surveillance Program,'' to incorporate three exceptions to Regulatory Guide (RG) 1.35, Revision 2, 1976. The exceptions concern the number of tendons detensioned, inspection of concrete adjacent to vertical tendons, and the time during which areas adjacent to tendons are inspected.

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?

No. The proposed change only clarifies TS requirements for the containment tendon surveillance program. The proposed clarification has been previously reviewed and approved by the NRC staff with Amendments 23 and 4, and is consistent with current regulatory guidance. As such, the proposed change is essentially administrative in nature. The containment tendon surveillance program has no impact on the probability of any accident initiators, and it will continue to ensure containment structural integrity. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

No. The proposed change only clarifies TS requirements for the containment tendon surveillance program. The proposed clarification has been previously reviewed and approved by the NRC staff with Amendments 23 and 4, and is consistent with current regulatory guidance. As such, the proposed change is essentially administrative in nature. Plant design and operation will not be changed, and no other safety related or important to safety equipment is affected by the proposed change. Therefore, the proposed change will 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.

No. The proposed change only clarifies TS requirements for the containment tendon surveillance program. The proposed clarification has been previously reviewed and approved by the NRC staff with Amendments 23 and 4, and is consistent with current regulatory guidance. As such, the proposed change is essentially administrative in nature. The containment prestressing system will continue to perform its function to ensure containment structural integrity. 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. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

Tennessee Valley Authority, Docket No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: June 25, 1999 (TS 98–016).

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) and TS Bases to reflect application of the Westinghouse generic Best Estimate Large Break Loss-of-Coolant Accident Analysis methodology using the WCOBRA/TRAC computer code.

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:

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

The proposed changes involve use of the Best Estimate Large Break loss-of-coolant accident (LOCA) analysis methodology and associated technical specification changes. Accumulator water level set points will be revised from [greater than or equal to] 7717 gallons and [less than or equal to] 8004 gallons to [greater than or equal to] 7630 gallons and [less than or equal to] 8000 gallons to provide the plant with an increased operating range. The plant conditions assumed in the analysis, including the accumulator water level instrumentation changes, are bounded by the design conditions for all equipment in the plant.

Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since it is shown that the emergency core cooling system (ECCS) is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. The small break LOCA analysis assumes only a nominal accumulator water level which is the same nominal value assumed in this analysis, therefore, the small break LOCA analysis is unaffected by the increase in the accumulator range. Also, the increased safety analysis range in accumulator water volume (+/−15 cubic feet) has an insignificant effect on the containment related analyses.

The post-LOCA containment sump boron calculation assumes a minimum accumulator volume which bounds (is smaller than) the 1005 cubic feet (7518 gallons) value supported by the Best Estimate Large Break LOCA analysis. Also, the hot leg switchover calculation models a maximum accumulator volume which is not bounded by the 1095 cubic feet (8191 gallons) maximum value supported by the Best Estimate Large Break LOCA analysis. However, an evaluation concludes that the Watts Bar hot leg switchover time is unaffected by the difference in maximum volumes.

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

No new modes of plant operation are being introduced by the new analysis or by the changes in instrumentation setpoints for accumulator water level. The parameters assumed in the analysis are within the design limits of existing plant equipment. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the WBN Unit 1 reactor system during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. The physical setpoint changes to accumulator water level instrumentation are bounded by the uncertainty evaluation addressing accumulator water level. A sufficient number of loss of coolant accidents with different break sizes, different locations, and other variations in properties have been considered to provide assurance that the most severe postulated LOCAs were evaluated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph b are met.

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 l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: December 14, 1999.

Description of amendment request: This proposed change would revise the Vermont Yankee (VY) Technical Specifications (TS) by relocating the procedural details of the Radiological Effluent Technical Specifications (RETS) to the Offsite Dose Calculation Manual (ODCM). The TS would also be revised to relocate procedural details associated with solid radioactive wastes to the Process Control Program (PCP). In addition, the TS definition for ''solidification'' would be relocated to the VY Technical Requirements Manual (TRM).

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 operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components to perform their intended safety function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents.

Implementation of programmatic controls for RETS already in TS will assure that the applicable regulatory requirements pertaining to the control of radioactive effluents will continue to be maintained. Since there are no changes to previous accident analysis, the radiological consequences associated with these analyses remain unchanged.

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

2. The operation of Vermont Yankee Nuclear Power Station 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 changes do not alter the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated. The proposed changes have no impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents.

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

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

There is no impact on equipment design or operation, and there are no changes being made to the TS-required safety limits or safety system settings that would adversely affect plant safety as a result of the proposed changes. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents. A comparable level of administrative controls will continue to be applied to those specifications being relocated to the ODCM, PCP, or TRM.

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: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: October 28, 1999, as supplemented December 21, 1999.

Description of amendment request: The proposed changes will remove the operability and surveillance requirements of Technical Specifications (TS) Section 3/4.6.4.3, ''Waste Gas Charcoal Filter System'' from the TS and relocate them to the Technical Requirements Manual.

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:

A waste gas decay tank rupture is highly unlikely, as the waste gas decay tanks are designed and constructed to stringent quality control standards, are provided with pressure relief valves to prevent overpressurization, are missile-shielded by installation below grade, and have their gaseous contents controlled to prevent potentially explosive mixtures. The entire gaseous content of the waste gas decay tank is assumed to be released to the atmosphere as a ground-level release * * *. The waste gas charcoal filter system is not credited for any mitigation of the release in the accident analysis for a waste gas decay tank rupture. In addition, the releases associated with a waste gas decay tank rupture are bounded by the existing LOCA [loss-of-coolant accident] releases. Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specification changes will not:

[1.] Involve a significant increase in the probability or consequences of an accident previously evaluated.

Relocating the operability and surveillance requirements for The Waste Gas Charcoal System to the TRM [Technical Requirements Manual] does not change the operation of the plant. The plant and the radioactive gas waste system will not be operated differently. No new accident initiators are established as a result of the proposed changes. Therefore, the probability of occurrence is not increased for any accident previously evaluated.

Relocating the operability and surveillance requirements for The Waste Gas Charcoal Filter to the TRM does not [a]ffect the gaseous releases to the environment, which are controlled by the ODCM [Offsite Dose Calculation Manual]. Additionally, no credit for these filters is taken in the accident analysis for Waste Gas Decay Tank rupture. Therefore, there is no increase in the consequences of any accident previously analyzed.

[2.] Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not affect the operation of the plant. The gaseous waste systems will not be operated differently as a result of the proposed changes. No new accident or event initiators are created moving the operability and surveillance requirements for The Waste Gas Charcoal Filter to the TRM. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type.

[3.] Involve a significant reduction in the margin of safety as defined in the bases on any Technical Specifications.

The proposed changes have no effect on any safety analyses assumptions. The waste gas charcoal filters are not used to mitigate the consequence[s] [of] a Waste Gas Decay Tank rupture. The accident analysis assumes total release of the radioactiv[ity] in the Waste Gas Decay Tank in the accident analysis. Therefore, the proposed changes do not result in a 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 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. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch Jr.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Dated of amendment request: November 29, 1999.

Description of amendment request: The proposed changes will modify the Technical Specifications in Section 4.7.7.1 for the Control Room Emergency Habitability System and Section 4.7.8.1 for the Safeguards Area Ventilation System. The changes will require laboratory testing of the charcoal filter carbon to be consistent with American Society for Testing and Materials (ASTM) Standard D3803–1989.

Basis for proposed no significant hazards consideration determination: In 10 CFR 50.92 three criteria are provided to determine whether a proposed license amendment involves a significant hazards consideration. No significant hazards consideration is involved if operation of the facility 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. 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:

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved. The proposed Technical Specification changes adopt the nuclear-grade charcoal testing requirements of ASTM D3803–1989 and do not affect the design or operation of the plant. The changes also do not involve any physical modification to the plant or result in a change in a method of system operation. The adoption of the 1989 edition of ASTM D3803 provides assurance that testing of nuclear-grade activated charcoal of ventilation systems is being performed with a suitable standard to ensure that charcoal adsorbers are capable of performing their required safety function and that the regulatory requirements regarding onsite and offsite dose consequences continue to be satisfied. The changes do not create an unreviewed safety question.

(a) The proposed changes modify surveillance testing requirements and do not affect plant systems or operation and therefore do not increase the probability or the consequences of an accident previously evaluated. The proposed surveillance requirements adopt ASTM D3803–1989 as the laboratory method for testing samples of the charcoal adsorber in response to NRC's Generic Letter 99–02. This method of testing charcoal adsorbers has been approved by the NRC as an acceptable method for determining methyl iodide removal efficiency. Since the charcoal adsorbers are used to mitigate the consequences of an accident, the more accurate the test, the better assurance we have that we remain within our accident analysis assumptions. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle. There is no change in the method of plant operation, system performance or system design. (b) The proposed changes do not create the possibility of an accident or malfunction of a different type than any evaluated previously.

The proposed changes modify surveillance testing requirements and do not impact plant systems or operations and therefore do not create the possibility of an accident or malfunction of a different type than evaluated previously. The proposed surveillance requirements adopt ASTM D3803–1989 as the laboratory method for testing samples of the charcoal adsorber. This change is in response to NRC's request in response to their Generic Letter 99–02. There is no change in the method of plant operation or system design. There are no new or different accident scenarios, transient precursors, nor failure mechanisms that will be introduced.

(c) The proposed changes modify surveillance test requirements and do not impact plant systems or operations and therefore do not significantly reduce the margin of safety. The revised surveillance requirements adopt ASTM D3803–1989 as the laboratory method for testing samples of the charcoal adsorber. The 1989 edition of this standard imposes very stringent requirements for establishing the capability of new and used activated carbon to remove radio-labeled methyl iodide from air and gas streams. The results of this test provide a more conservative estimate of the performance of nuclear-graded activated carbon used in all nuclear power plant HVAC [heating, ventilation, and air conditioning] systems for the removal of radioiodine. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle.

This analysis demonstrates that the proposed amendment to The North Anna Units 1 and 2 Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and 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 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. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch Jr.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed no Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Dated of amendment request: April 11, 1996, as supplemented April 6, 1998, March 22 and July 29, 1999 (PCN–460).

Brief description of amendment request: The proposed amendments would revise the San Onofre Units 2 and 3 Technical Specification (TS) related to the containment isolation valves. Specifically, the licensee proposed a revision to TS 3.6.3 to extend the completion times for Section D.1 and D.2 valves from 4 hours to the applicable limiting condition for operation time pertaining to the engineered safety feature system in which the valve is installed.

Dated of publication of individual notice in **Federal Register:** January 19, 2000 (65 FR 2993), as corrected January 26, 2000 (65 FR 4265).

Expiration date of individual notice: February 18, 2000.

Southern California Edison Company, et al., Docket Nos.50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment request: January 2, 1998, as supplemented December 13, 1999 (PCN–482).

Brief description of amendment request: The proposed amendments would revise the San Onofre Units 2 and 3 Technical Specification (TS) relating to the Auxiliary Feedwater (AFW) System. Specifically, the licensee proposed to revise TS 3.7.5 to add a note that states: ''The steam driven AFW pump is OPERABLE when running and controlled manually to support plant start-ups, plant shut-downs, and AFW pump and valve testing.''

Date of publication of individual notice in **Federal Register***:* January 19, 2000 (65 FR 2991), as corrected January 26, 2000 (65 FR 4265).

Expiration date of individual notice: February 18, 2000.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment request: January 11, 1999, as supplemented November 29, 1999 (PCN–499).

Brief description of amendment request: The proposed amendments would revise the San Onofre Units 2 and 3 Technical Specification (TS) 3.7.6, ''Condensate Storage Tank (CST T–121 and T–120)'' to change the minimum inventory of water maintained in the condensate storage tank (T–120) from 280,000 gallons to 360,000 gallons during plant operation Modes 1, 2 and 3.

Date of publication of individual notice in **Federal Register***:* January 18, 2000 (65 FR 2648).

Expiration date of individual notice: February 17, 2000.

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, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of application for amendment: September 28, 1999.

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirement 3.7.6.2 ''Component Cooling Water (CCW) System'' to change the CCW pump automatic start actuation signal basis from Engineered Safety Feature Actuation Signal to Loss-of-Power Diesel Generator.

Date of issuance: January 21, 2000.

Effective date: January 21, 2000.

Amendment No.: 186.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register***:* November 3, 1999 (64 FR 59798).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 21, 2000.

No significant hazards consideration comments received: No.

Consumers Energy Company, Docket No. 50–155, Big Rock Point Plant, Charlevoix County, Michigan

Date of amendment request: May 11, 1999, as supplemented June 3 and July 28, 1999.

Brief description of amendment: This amendment deletes from the Defueled Technical Specifications (DTS) subsection 1.16, ''SITE BOUNDARY,'' Figure 5.1–1, the Big Rock Point (BRP) Site Map, and DTS 5.1.1 paragraph numbering and removes certain site-specific information from DTS 5.1, which describes the BRP site. The amendment also makes editorial changes to DTSs 6.6.2.5. g, h, and j, and 6.6.2.6.b. because of the changes associated with DTSs 1.16 and 5.1 and Figure 5.1–1 described above. Most of the information removed or deleted from the DTSs can be found in the BRP Final Hazards Summary Report.

Date of Issuance: January 13, 2000.

Effective Date: January 13, 2000, to be implemented within 45 days from date of issuance. Implementation includes incorporation of the site boundary information, as discussed in the staff's safety evaluation enclosed with this amendment, into the next Final Safety Analysis Report (*i.e.* the updated Final Hazards Summary Report for the Big Rock Point Nuclear Plant) update in accordance with the schedule in 10 CFR 50.71(e).

Amendment No.: 121.

Facility Operating License No. DPR–6. The amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** June 16, 1999 (64 FR 32288).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 13, 2000.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: April 5, 1999 as supplemented by letters dated May 27, July 6, October 7, and November 22, 1999.

Brief description of amendments: The amendments revised the Technical Specifications to incorporate Topical Report DPC–NE–3005–P, ''Thermal Hydraulic Transient Analysis Methodology.''

Date of Issuance: January 18, 2000.

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

Amendment Nos.: Unit 1–309; Unit 2–309; Unit 3–309.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 30, 1999 (64 FR 35202), November 3, 1999 (64 FR 59801).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 18, 2000.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina.

Date of application of amendments: September 29, 1999.

Brief description of amendments: The amendments revise the Containment Inservice Inspection Program Technical Specifications related to the containment leakage testing program and the pre-stressed concrete containment tendon surveillance program.

Date of Issuance: January 18, 2000.

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

Amendment Nos.: Unit 1—310; Unit 2—310; Unit 3—310.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 17, 1999 (64 FR 62707).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 18, 2000.

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 16, 1999.

Brief description of amendment: The license amendment revises the River Bend Station Technical Requirements Manual, Section TR 3.9.14, and adds a temporary exception to the current prohibition for travel of loads in excess of 1200 pounds over fuel assemblies in the spent fuel storage pool. The exception allows the licensee to move the spent fuel pool (SFP) watertight gates, which separate the SFP from the cask and lower transfer pools, in order to perform repairs on the gates and watertight seals prior to the end of Refueling Outage 9. Updated Safety Analysis Report (USAR) Sections 9.1.2.2.2 and 9.1.2.3.3 are also changed to reflect the proposed exception.

Date of issuance: January 13, 2000.

Effective date: The license amendment is effective as of its date of issuance and shall be implemented in the next periodic update to the USAR and TRM in accordance with 10 CFR 50.71(e). Implementation of the amendment is the incorporation into the USAR and TRM update, the changes to the description of the facility as described in the licensee's application dated December 16, 1999, as supplemented by letters dated December 21, 1999, and January 10, 2000, and evaluated in the staff's Safety Evaluation attached to this amendment.

Amendment No.: 108.

Facility Operating License No. NPF–47: The amendment revises the Technical Requirements Manual and Updated Safety Analysis Report.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (64 FR 71511 dated December 21, 1999). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by January 20, 2000, but indicated that if the Commission made a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final NSHC determination are contained in a Safety Evaluation dated January 13, 2000.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 13, 2000.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: December 16, 1999

Brief description of amendment: The proposed change would amend Technical Specification 4.18.5.b to allow tube 110/60 to remain in service through the current operating cycle (cycle 16) with two axial indications that have potential through-wall depths greater than the plugging limit. The axial indications are located in the roll transition region and are contained within the upper tubesheet.

Date of issuance: January 13, 2000.

Effective date: As of the date of issuance.

Amendment No.: 203.

Facility Operating License No. DPR–51: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (64 FR 73080 dated December 29, 1999). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by January 28, 2000, 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, finding of exigent circumstances, and final NSHC determination are contained in a Safety Evaluation dated January 13, 2000.

Attorney for Licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc.,Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: July 20, 1998, as supplemented by letter dated June 29, 1999.

Brief description of amendment: The amendment incorporates the Technical Specification changes necessary for implementation of the Boiling Water Reactor Owners' Group Reactor Stability Long-Term Solution, Enhanced Option 1–A.

Date of issuance: January 19, 2000.

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

Amendment No: 141.

Facility Operating License No. NPF–29: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register***:* August 25, 1999 (64 FR 46432).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 19, 2000.

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: October 12, 1999.

Brief description of amendments: The amendments would revise the Technical Specifications (TSs) Surveillance Requirement 4.6.2.2.d for the spray additive system to relocate the details associated with the acceptance criteria and test parameters to the associated TSs Bases. Additionally, certain administrative text format changes were made.

Date of issuance: January 19, 2000.

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

Amendment Nos.: 240 and 221.

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

Date of initial notice in **Federal Register:** November 3, 1999 (64 FR 59804).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 19, 2000.

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 11, 1998, as supplemented by letters dated January 14, and August 5, 1999.

Brief description of amendments: The amendments revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise TS 6.8.4f., ''Containment Polar and Turbine Building Cranes,'' to control the operation of the containment polar cranes in jet impingement zones during Modes 1, 2, 3, and 4.

Date of issuance: January 12, 2000.

Effective date: January 12, 2000.

Amendment Nos.: Unit 1–137; Unit 2–137.

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

Date of initial notice in **Federal Register:** April 21, 1999 (64 FR 19561).

The August 5, 1999, supplemental letter 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 January 12, 2000.

No significant hazards consideration comments received: No.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: September 29, 1999, as supplemented December 7, 1999.

Brief description of amendment: This amendment revises the Technical Specifications to allow, on a one-time basis only, the Power Authority of the State of New York to extend the allowed out-of-service time for the Residual Heat Removal Service Water (RHRSW) System from 7 days to 11 days. This amendment is only applicable during installation of Modification 99–095 to the ''A'' RHRSW Strainer.

Date of issuance: January 28, 2000.

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

Amendment No.: 259.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 20, 1999 (64 FR 56532).

The December 7, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 28, 2000.

No significant hazards consideration comments received: No.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: August 25, 1999.

Brief description of amendments: The amendments authorize the licensee to perform single-cell charging of operable safety-related batteries by using non-Class 1E single-cell battery chargers, with proper electrical isolation. The single-cell chargers would be used to restore individual cell float voltage to the normal TS limit.

Date of issuance: January 24, 2000.

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

Amendment Nos.: 226 and 207.

Facility Operating License Nos. DPR–70 and DPR–75.: Amendments revised the licenses.

Date of initial notice in **Federal Register:** September 22, 1999 (64 FR 51349).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 24, 2000.

No significant hazards consideration comments received: No.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: June 28, 1999.

Brief description of amendment: The amendment revises the Ginna Station Improved Technical Specifications associated with the Reactor Coolant System Leakage Detection Instrumentation.

Date of issuance: January 19, 2000.

Effective date: January 19, 2000.

Amendment No.: 76.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 11, 1999 (64 FR 43778)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 19, 2000.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: April 19, 1999, as supplemented by letter dated November 1, 1999.

Brief description of amendments: The amendments revised the Technical Specifications Surveillance Requirement (SR) 3.3.5.2 and associated Bases to allow the loss of voltage and degraded voltage trip setpoints to be treated as ''nominal'' values.

Date of issuance: January 19, 2000

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

Amendment Nos.: 111—Unit 1; 89—Unit 2.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 1, 1999 (64 FR 67340).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 19, 2000.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: September 8, 1999, as supplemented November 9, 1999.

Brief description of amendments: The amendments revised Technical Specification 3/4.8.1, ''A.C. Sources, Operating,'' and associated Bases, by deleting the 18-month surveillance to subject the standby diesel generator to inspections in accordance with procedures prepared in conjunction with its manufacturer's recommendations. The surveillance requirements have been relocated to the Technical Requirements Manual.

Date of issuance: January 14, 2000.

Effective date: January 14, 2000, to be implemented within 30 days.

Amendment Nos.: Unit 1—121 ; Unit 2—109

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 1, 1999 (64 FR 67341).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 14, 2000.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: September 8, 1999, as supplemented by letter dated November 9, 1999.

Brief description of amendments: The amendments revised Technical Specification (TS) 3/4.8.1, ''A.C. Sources, Operating,'' and associated Bases, by eliminating the requirement for accelerated testing of the standby diesel generators and the associated reporting requirements. The TS Index was also revised to reflect these changes.

Date of issuance: January 14, 1999.

Effective date: January 14, 1999.

Amendment Nos.: Unit 1—122 ; Unit 2—110.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 3, 1999 (64 FR 59806).

The November 9, 1999, supplement provided additional clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 14, 2000.

No significant hazards consideration comments received: No

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: October 21, 1999.

Brief description of amendment: The amendment corrects two textual errors and changes the designation of a referenced figure.

Date of Issuance: January 11, 2000.

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

Amendment No.: 183.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 17, 1999 (64 FR 62717).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 11, 2000.

For the Nuclear Regulatory Commission.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 2nd day of February 2000.

For the Nulcear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–2835 Filed 2–8–00; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

Proposed Collections; Comment Request

Upon Written Request; Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension:

Form S–2, SEC File No. 270–60, OMB Control No. 3235–0072
Form F–1, SEC File No. 270–249, OMB Control No. 3235–0258
Form F–2, SEC File No. 270–250, OMB Control No. 3235–0257
Form F–3, SEC File No. 270–251, OMB Control No. 3235–0256
Form F–7, SEC File No. 270–331, OMB Control No. 3235–0383
Form F–8, SEC File No. 270–332, OMB Control No. 3235–0378
Form F–X, SEC File No. 270–336, OMB Control No. 3235–0379
Form 10–SB, SEC File No. 270–367 OMB Control No. 3235–0419
Form DF, SEC File No. 270–430, OMB Control No. 3235–0482
Form T–1, SEC File No. 270–121, OMB Control No. 3235–0110
Form T–2, SEC File No. 270–122, OMB Control No. 3235–0111
Form T–3, SEC File No. 270–123, OMB Control No. 3235–0105
Form T–4, SEC File No. 270–124, OMB Control No. 3235–0107
Schedule 13E–4F, SEC File No. 270–340, OMB Control No. 3235–0375
Schedule 14D–1F, SEC File No. 270–338, OMB Control No. 3235–0376
Schedule 14D–9F, SEC File No. 270–339, OMB Control No. 3235–0382
Rule 14f–1, SEC File No. 270–127, OMB Control No. 3235–0108
Rule 12d1–3, SEC File No. 270–116, OMB Control No. 3235–0109

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*) the Securities and Exchange Commission (Commission) is soliciting comments on the collections of information summarized below. The Commission plans to submit these existing collections of information to the Office of Management and Budget for approval.

Form S–2 is used for registration of securities of certain issuers. The Form S–2 provides investors with the necessary information to make investment decisions regarding securities offered to the public. The likely respondents will be public companies. The information collected must be filed with the Commission. All information is provided to the public upon request. Form S–2 takes 470 burden hours to prepare and is filed by 101 respondents for a total of 47,470 burden hours.

Form F–1 is a registration statement of securities of certain foreign private issuers. Form F–1 provides the public with the necessary information to make informed investment decisions regarding securities offered to the public by foreign private issuers. The information provided on Form F–1 is mandatory. All information on Form F–1 is reported to the public upon request. Form F–1 takes approximately 1,868 burden hours to prepare and is filed by 170 respondents. It is estimated that 25% of the 317,560 total burden hours (79,390 hours) would be prepared by the company.

Form F–2 is a registration statement of securities of certain foreign private issuers. Form F–2 provides the public with the necessary information to make