charges. ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at *http://www.nrc.gov/ACRSACNW.*

(g) Video teleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician, (301–415–8066) between 7:30 a.m. and 3:30 p.m. Eastern Time at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the video teleconferencing link. The availability of video teleconferencing services is not guaranteed.

ACRS Subcommittee Meetings

ACRS Subcommittee meetings will also be conducted in accordance with the above procedures, as appropriate. When Subcommittee meetings are held at locations other than at NRC facilities, reproduction facilities may not be available at a reasonable cost. Accordingly, 25 additional copies of the materials to be used during the meeting should be provided for distribution at such meetings.

Special Provisions When Proprietary Sessions Are To Be Held

If it is necessary to hold closed sessions for the purpose of discussing matters involving proprietary information, persons with agreements permitting access to such information may attend those portions of the ACRS meetings where this material is being discussed upon confirmation that such agreements are effective and related to the material being discussed.

The Designated Federal Official should be informed of such an agreement at least five working days prior to the meeting so that it can be confirmed, and a determination can be made regarding the applicability of the agreement to the material that will be discussed during the meeting. The minimum information provided should include information regarding the date of the agreement, the scope of material included in the agreement, the project or projects involved, and the names and titles of the persons signing the agreement. Additional information may be requested to identify the specific agreement involved. A copy of the executed agreement should be provided to the Designated Federal Official prior to the beginning of the meeting for admittance to the closed session.

Dated: September 28, 2001.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 01–24704 Filed 10–2–01; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

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 September 10, 2001 through September 21, 2001. The last biweekly notice was published on September 19, 2001 (66 FR 48283).

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

By November 2, 2001, 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 NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* 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: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: June 21, 2001 (U–603490)

Description of amendment request: The proposed amendment would eliminate the Technical Specification leakage limit for any one main steam line as measured by main steam isolation valve leakage of less than or equal to 28 standard cubic feet per hour (scfh) and replace that requirement with an aggregate leakage limit of less than or equal to 112 scfh for all four main steam lines.

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:

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

The MSIVs [main steam isolation valves] are not initiators of or precursors to any of the accident scenarios presented in the Updated Safety Analysis Report. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

The proposed change to the TS [Technical Specifications] modifies the allowed main steam line leakage limit to an aggregate value (i.e., leakage for all four main steam lines combined) with no change to the currently allowed total leakage rate. This is the value currently used for calculation of dose consequences for the bounding accident for which MSIV closure is credited, the large-break loss of coolant accident (LOCA). This proposed change does not impact or increase the assumed radionuclide source term therefore; this change does not involve an increase in consequences of any accident previously evaluated.

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

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

The proposed change does not involve a change to the plant design or operation. No new equipment will be installed or utilized, and no new operating conditions will be initiated as a result of this change. The safety function of the MSIVs is to provide timely steam line isolation to mitigate the release of radioactive steam and limit reactor inventory loss under certain accident and transient conditions. Changing the leakage limits to include an aggregate value does not affect the isolation function performed by the MSIVs. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The total allowed leakage rate for all four main steam lines remains unchanged at ≤112 scfh. The proposed change does not challenge the integrity of the fuel cladding, reactor coolant pressure boundary, or the primary containment.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage. As stated above, the total allowed leakage rate for all four main steam lines remains unchanged. In addition, there will not be a change in the types or amounts of any effluents released offsite. The radiological analyses remain unchanged and within the guidelines of 10 CFR 100, ''Reactor Site Criteria,'' and 10 CFR 50, ''Domestic Licensing of Production and Utilization Facilities,'' Appendix A, ''General Design Criteria for Nuclear Power Plants,'' General Design Criterion 19, ''Control Room.''

Therefore, the proposed 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: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, IL 60515.

NRC Section Chief: Anthony J. Mendiola.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: June 21, 2001 (U–603495).

Description of amendment request: The proposed amendment would modify the Technical Specification requirement that the main steam line safety relief valves (SRVs) open when they are manually actuated by instead requiring that the SRV valve actuators stroke on a manual actuation.

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:

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

The proposed changes modify TS [Technical Specification] SR [surveillance requirements] 3.4.4.3, SR 3.5.1.7 and SR 3.6.1.6.1. The proposed changes will eliminate the TS requirement that each valve opens during the manual actuation of the SRVs [safety relief valves]. Accidents are initiated by the malfunction of plant equipment, or the catastrophic failure of plant structures, systems or components. The performance of SRV testing is not a precursor to any accident previously evaluated and does not change the manner in which the SRVs are operated. The proposed testing requirements will not contribute to the failure of the SRVs nor any plant structure, system or component. Thus, the proposed changes to the performance of SR 3.4.4.3, SR 3.5.1.7 and SR 3.6.1.6.1 do not have any affect on the probability of an accident previously evaluated.

The performance of SRV testing provides assurance that the SRVs are capable of depressurizing the reactor pressure vessel (RPV). This will protect the reactor vessel from overpressurization and allowing the combination of the Low Pressure Coolant Injection (LPCI) System and Low Pressure Core Spray (LPCS) System to inject into the RPV as designed. The LLS [low-low set] logic causes two LLS valves to be opened at a lower pressure than the relief or safety mode pressure setpoints and causes all the LLS valves to stay open longer, such that reopening of more than one SRV is prevented on subsequent actuations. Thus, the LLS function prevents excessive short duration SRV cycles with valve actuation at the relief setpoint. The proposed changes involve the manner in which the subject valves are tested, and have no affect on the types or amounts of radiation released or the predicted offsite doses in the event of an accident. The proposed testing requirements are sufficient to provide confidence that the SRVs, ADS valves and the LLS valves will perform their intended safety functions. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed changes to SR 3.4.4.3, SR 3.5.1.7 and SR 3.6.1.6.1 do not affect the assumed accident performance of the SRVs, nor any plant structure, system or component previously evaluated. The proposed changes do not install any new equipment, and installed equipment is not being operated in a new or different manner. The valves continue to be bench-tested to verify the safety and relief modes of valve operation. The proposed changes will allow the testing of the manual actuation electrical circuitry, solenoid and air control valve, and the actuator without causing the SRV to open. No setpoints are being changed which would alter the dynamic response of plant equipment. Administrative controls, such as verifying that the actuator assembly has been recoupled following testing, minimize the potential for valve failures. Accordingly, no new failure modes are introduced. The changes credit the performance of bench testing, setpoint verification and in-situ actuator exercising with providing sufficient testing to ensure the valves will perform their required safety functions.

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

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

The proposed changes to SR 3.4.4.3, SR 3.5.1.7 and SR 3.6.1.6.1 will allow the uncoupling of the SRV stem from the other components associated with the manual actuation of the SRVs. The proposed changes will allow the testing of the manual actuation electrical circuitry, solenoid and air control valve, and the actuator without causing the SRV to open. The SRVs will continue to be manually actuated by the bench-test valve control system of the setpoint testing program and prior to installation in the plant. The proposed changes do not effect the valve setpoint or the operational criteria that directs the SRVs to be manually opened during plant transients. There are no changes proposed which alter the setpoints at which protective actions are initiated, and there is no change to the operability requirements for equipment assumed to operate for accident mitigation.

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

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

Attorney for licensee: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, IL 60515.

NRC Section Chief: Anthony J. Mendiola.

Duke Energy Corporation, Docket No. 50–287, Oconee Nuclear Station, Unit 3, Oconee County, South Carolina

Date of amendment request: March 5, 2001, supplemented September 4, 2001

Description of amendment request: The proposed amendments would revise the Technical Specifications for Unit 3 to allow a one-time extension of the 10 CFR Part 50, Appendix J Containment Integrated Leak Rate Test interval. Presently the 10-year interval test is required to be performed prior to the operating cycle before the outage when the steam generators will be replaced. The proposed amendment would extend the test approximately 16 months to the outage when they will be replaced (i.e., no later than April 11, 2005), thereby precluding the need to perform the test during two subsequent outages. The No Significant Hazards Consideration Determination contained in the March 5, 2001, submittal was superceded in the September 4, 2001, submittal and is presented below.

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

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

Response: No

The proposed revision to the Oconee Nuclear Station, Unit 3 (ONS–3) Technical Specifications (TS) adds a one-time extension to the current interval for Type A testing (containment Integrated Leak Rate Testing (ILRT)). The current test interval of 10 years, would be extended on a one time basis to 12 years 7 months from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A testing extension is not a modification to plant systems, or a change to plant operation that could initiate an accident. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493 found that, generically, very few potential containment leakage paths fail to be identified by Type B and C tests. In fact, an analysis of 144 ILRT results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A testing frequency to one per twenty years would lead to an imperceptible increase in risk. The NUREG conclusions are supported by an ONS–3 specific evaluation of risk and consequences. ONS–3 provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. Inspections required by the Maintenance Rule and American Society of Mechanical Engineers (ASME) code are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing required by the ONS–3 TS will identify any containment opening, such as valves, that would otherwise be detected by the Type A tests. Type B and C testing is performed at the frequency specified by 10 CFR 50, Appendix J, Option A, § D.2 and § D.3, respectively. These factors show that a ONS–3 Type A test extension will not represent a significant increase in the consequences of an accident.

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

Response: No

The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes [b]eing made to the plant. There are no changes to the operation of the plant that could introduce a new failure mode creating the possibility of a new or different kind of accident.

3. Do the proposed changes involve a significant reduction in the margin of safety?

Response: No

The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20 year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes a very small amount to the individual risk, and that the decrease in Type A testing frequency would have a minimal affect on this risk since most potential leakage paths are detected by Type C testing. The NUREG conclusions are supported by an ONS–3 specific evaluation of risk and consequences.

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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard L. Emch, Jr.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: April 24, 2001.

Description of amendment request: The proposed amendment would revise the Indian Point 3 (IP3) Final Safety Analysis Report (FSAR) to reflect the original plant design. It will indicate that a portion of one loop of the Component Cooling Water (CCW) System is routed in the non-safety-related portion of the Fuel Storage Building (FSB).

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 Indian Point 3 plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

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

The proposed change to the FSAR revises it to reflect the as built configuration of the Component Cooling Water (CCW) system. One loop is routed to the Spent Fuel Pit Heat Exchanger (SFPHX) located in the Fuel Storage Building (FSB). The portion of the FSB where the CCW is located is seismic Class III rather than the seismic Class I required by design criteria in the FSAR. The proposed change demonstrates that the CCW loop and SFPHX will not be affected by a seismic event and that operator, action with credit for the Primary Water Storage Tank (PWST) providing redundancy (a source of water to maintain CCW), will assure that the CCW system function can be performed following a tornado. The proposed change does not affect the probability of an accident previously evaluated because there is no design change and the probability of natural phenomena does not change. The proposed change does not affect the consequences of an accident previously evaluated because the CCW system function is maintained by operator action following a tornado with missile damage to a small bore CCW pipe.

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

The proposed change to the FSAR revises it to reflect the as built configuration of the CCW system. One loop is routed to the Spent Fuel Pit Heat Exchanger (SFPHX) located in the Fuel Storage Building (FSB). The portion of the FSB where the CCW is located is seismic Class III rather than the seismic Class I required by design criteria in the FSAR. The proposed change demonstrates that the CCW loop and SFPHX will not be affected by a seismic event and that operator action with credit for the PWST inventory (a source of water to maintain CCW) will assure that the CCW system function can be performed following a tornado. The proposed change does not create the possibility of a new or different kind of accident because the CCW system will not be operated differently than designed and operator action and the use of components to perform redundant functions to cope with a tornado is currently approved in the FSAR. Also, the CCW is designed to be operated by separating the loops.

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

The proposed change to the FSAR revises it to reflect the as built configuration of the CCW system. One loop is routed to the Spent Fuel Pit Heat Exchanger (SFPHX) located in the Fuel Storage Building (FSB). The portion of the FSB where the CCW is located is seismic Class III rather than the seismic Class I required by design criteria in the FSAR. The proposed change demonstrates that the CCW loop and SFPHX will not be affected by a seismic event and that operator action with credit for the PWST inventory (a source of water to maintain CCW) will assure that the CCW system function can be performed following a tornado. The proposed change does not involve a significant reduction in a margin of safety because the existing plant design considers the use of operator action and redundant components to mitigate the effects of a tornado. Also, the proposed change is for damage caused by a low risk event. The risk of tornado damage to the CCW piping in the FSB is low. The IP3 examination of external events found the probability of any tornado striking IP3 to be 1.59E–4/year. For tornados with wind speeds in excess of 180 mph, the frequency decreases to 8.62E–7/year. For the design basis tornado with a 300 mph wind speed, the frequency is 1.02E–9/year. The risk of a tornado following a LOCA is lower. The frequency of a LOCA followed by any tornado within 30 days is 3.02E–8/year. The frequency of the event can be used as a conservative estimate of core damage frequency (CDF). When compared to the nominal CDF at IP3, the frequency is negligible.

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

Attorney for licensee: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Generating Station, 600 Rocky Hill Road, Plymouth, MA 02360.

NRC Section Chief: Peter Tam, Acting.

Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois; Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois; Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois; Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: March 23, 2001.

Description of amendment request: The proposed amendment would incorporate changes to the Physical Security Plan and Guard Force Training and Qualification Plans for the identified facilities. The proposed changes would modify current escorting and control requirements for non-designated vehicles, lighting requirements for exterior areas within the protected area, and annual weapons qualifications.

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

No physical plant changes are being made as a result of changing the vehicle, lighting, and weapons qualification requirements. The proposed changes involve revising requirements that provide little or no value in the protection of the facility with regards to the design basis threat as described in 10 CFR part 73, ''Physical Protection of Plants and Materials,'' paragraph 1(a). Because the defensive strategies at each station have been proven to be effective without reliance on these requirements, it is concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no physical changes being made to the plant as a result of changing the vehicle, lighting, and weapons qualification requirements. The defensive strategies at each station remain unchanged under the proposed changes. A review of possible intrusion scenarios has confirmed that no event would result in a new sequence of events that could lead to a new accident scenario. Based on this review, it is concluded that no accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. Therefore, the proposed Physical Security Plan and Guard Force Training and Qualification Plan changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

It has been shown during recent Operational Safeguards Readiness Evaluations (OSRE), that the proposed changes do not impact the security's ability to protect the facility from the threat of radiological sabotage. The risk of radiological sabotage would not be increased by changing the vehicle, lighting, and weapons qualification requirements. Additionally, proposed change in weapons qualifications provides a more realistic evaluation of a responder's ability to protect the station from the threat of radiological sabotage. Based on this review, the proposed amendment does not involve a significant reduction in the margin of safety.

Therefore, based upon the above evaluation, Exelon Generation Company, LLC has concluded that these changes do not involve significant hazards considerations.

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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Jr., Vice President and General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, KSB 3-W, Kennett Square, PA 19348.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and 2), Beaver County, Pennsylvania

Date of amendment requests: May 22, 2001.

Description of amendment requests: The proposed amendments would revise the BVPS–1 and 2 technical specifications (TSs) to implement improvements endorsed in the Nuclear Regulatory Commission's (NRC's) Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132). These license amendment requests propose the addition of an administrative control program for explosive gas and storage tank radioactivity monitoring to the administrative controls section of the BVPS–1 and 2 TSs consistent with the corresponding standard TS program. The amendment requests propose to relocate the TS requirements associated with the curie content limit for liquid and gaseous waste storage system and the explosive gas concentration limit for gaseous waste storage systems. The addition of the standard TS program provides an appropriate level of control for the affected requirements in the TSs that allows these details to be relocated. The TSs proposed for relocation will be placed in the BVPS–1 and 2 Offsite Dose Calculation Manual or the BVPS–1 and 2 Licensing Requirements Manual. The net effect of the proposed changes is to provide adequate regulatory control in the TSs while making the content of the BVPS–1 and 2 TSs more consistent with the standard TSs for Westinghouse plants as presented in NUREG–1431 and simplifying the BVPS–1 and 2 TSs consistent with the goals of the NRC Final Policy Statement on TS improvements for nuclear power reactors.

Additionally, revisions to BVPS–1 and 2 TS 6.9.3, ''Annual Radioactive Release Report,'' are proposed to include changes to the reporting requirements. The changes proposed also include various administrative revisions to support the relocations.

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

The proposed amendment does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any event initiator. Nor is any analyzed accident scenario being revised. The initiating conditions and assumptions for accidents described in the UFSAR [Updated Final Safety Analysis Report] remain as previously analyzed.

The proposed amendment also does not involve a significant increase in the consequences of an accident previously evaluated. The amendment does not reduce the current operability requirements contained in the TS proposed for relocation. The proposed relocation of TS requirements only affects the level of regulatory control involved in future changes to the requirements. The proposed changes include additions to the TS in the form of programmatic controls that effectively replace the key TS requirements being relocated. As such, the TS proposed for relocation no longer meet the 10 CFR 50.36 criteria for retention in the TS.

The additional administrative changes are editorial in nature, and are made to support the relocation of TS. The additional administrative changes and the changes to Specification 6.9.3 have no adverse effect on the safety analyses for design basis accidents described in the UFSAR. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed.

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

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

The proposed amendment does not involve any physical changes to the plant or the modes of plant operation defined in the TS. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

There are no changes in this amendment that would cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new mode of failure has been created and no new equipment performance requirements are imposed. The proposed amendment has no effect on any previously evaluated accident.

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

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

The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this amendment revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

The proposed change includes the addition of programmatic controls that allow the affected TS to be relocated. The relocation of TS does not reduce the effectiveness of the requirements being relocated. Rather, the relocation of the TS results in a change in the regulatory control required for future changes made to the requirements. Additionally, due to the new programmatic controls, the TS proposed for relocation no longer meet the 10 CFR 50.36 criteria for retention in the TS.

The requirements contained within the affected TS will continue to be implemented by the appropriate plant procedures (e.g., operating and maintenance procedures) in the same manner as before. However, future changes to the relocated requirements will be controlled in accordance with 10 CFR 50.59 instead of a license amendment pursuant to 10 CFR 50.90. The provisions of 10 CFR 50.59 establish adequate controls over requirements removed from the TS and assure future changes to these requirements will be consistent with safe plant operation.

The additional administrative changes are editorial in nature, and are made to support the relocation of TS. The additional administrative changes and the proposed changes to Specification 6.9.3 do not alter any operating parameters or design requirements assumed in a safety analysis for systems or components important to the mitigation and control of design bases accident conditions within the facility. Nor do these changes alter safety limits or safety system settings required for safe operation of the plant or the assumptions of any safety analysis.

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

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

Attorney for licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Peter Tam (Acting).

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2 (BVPS–2), Beaver County, Pennsylvania

Date of amendment request: July 25, 2001.

Description of amendment request: The proposed license amendment for BVPS–2 would increase the limits for boron concentration in the refueling water storage tank (RWST) and in the reactor coolant system (RCS) accumulators. The RCS minimum boron concentration limit for Mode 6 would also be revised to make it consistent with the RWST boron concentration limit. The increase in the boron concentration limits in the RWST and accumulators is needed to address higher reactor core reactivity levels associated with core operation at higher plant capacity factors. TS Bases changes are also proposed to reflect the changes discussed above.

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

No. The proposed change to increase the boron concentration in the Beaver Valley Power Station (BVPS) Unit 2 Refueling Water Storage Tank (RWST), Accumulators and in the Reactor Coolant System (RCS) during Mode 6 will maintain the safety analyses results in Chapter 15 of the BVPS Unit 2 Updated Final Safety Analysis Report (UFSAR) as bounding values for all Loss Of Coolant Accident (LOCA) and non-LOCA design basis accidents. The proposed changes do not reduce the RWST or accumulators ability to meet their design bases, which will not result in a significant increase in the probability of an accident previously evaluated.

Increased boron concentration limits for the RWST, Accumulators, and RCS in Mode 6 will not increase the consequences of an accident previously analyzed as described in the UFSAR. The increased boron concentration limits reduce the time to switchover from cold leg to hot leg recirculation, which will prevent boron precipitation in the reactor vessel following a LOCA. The post-LOCA long term core cooling minimum boron requirements have been determined to continue to be adequate to ensure adequate post-LOCA shutdown margin. The post-LOCA containment sump and containment spray pH remain within the limits specified in the UFSAR. All other transients either were not impacted or were made less severe as a result of the increased boron concentrations.

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

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

No. The proposed increase in boron concentration does not add new or different equipment to the facility. The proposed Technical Specification changes also do not alter the manner in which plant equipment is being operated. Although the increased boron concentration requires procedure changes to ensure that cold leg to hot leg recirculation after a LOCA occurs quicker, there are no changes to the methods utilized to respond to plant events. The proposed Technical Specification changes do not alter instrument or control setpoints that initiate protective or mitigative actions. These increased boron concentration limits are conservative and do not alter the RCS or Emergency Core Cooling Systems' ability to perform their design bases.

Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated accident since the RCS will continue to operate in accordance with their design bases.

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

No. The LOCA considerations, including Peak Cladding Temperature calculations, containment sump and spray pH requirements, boron solubility requirements, cold shutdown boration requirements, post-LOCA long term core cooling minimum boron requirements, hot leg recirculation switchover requirements, post-LOCA hydrogen generation requirements, and radiological requirements have been evaluated and determined to be acceptable. The acceptance criteria of all non-LOCA design basis accidents continue to be met.

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not adversely affect the ability of systems, structures or components important to the mitigation and control of design bases accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling condition for extended periods of time.

Based upon the above evaluations, [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: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Peter Tam, Acting.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: August 22, 2001.

Description of amendment request: The proposed amendments revise Section 6.0, ''Administrative Controls,'' of the Technical Specifications (TS) to change the title of the corporate executive responsible for overall nuclear safety from ''President-Nuclear Division'' to ''Chief Nuclear Officer.'' The proposed changes eliminate the reference to a specific organizational title and replace it with a generic organizational position title. This conforms the TS to a recent organizational change and precludes the need for future amendments in response to future corporate title changes.

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) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments are administrative in nature, changing the title of the corporate executive responsible for overall plant nuclear safety in St. Lucie Units 1 and 2 TS, and would not involve a significant increase in the probability or consequences of an accident previously evaluated. These amendments do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect TS that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the TS are administrative in nature, changing the title of the corporate executive responsible for overall plant nuclear safety in St. Lucie Units 1 and 2 TS, and would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature, changing the title of the corporate executive responsible for overall plant nuclear safety in the St. Lucie Units 1 and 2 TS, and would not reduce any of the margins of safety.The operating limits and functional capabilities of the affected systems, structures, and components remain unchanged by the proposed amendments. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: August 15, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to extend the channel calibration surveillance frequency for the automatic depressurization system (ADS) timers from 18 months to 24 months. Specifically, SR 3.3.5.1.7 (18-month CHANNEL CALIBRATION) surveillance requirement listed in Table 3.3.5.1–1, functions 4.b. and 5.b. (ADS Timer), would be changed to SR 3.3.5.1.8 (24-month CHANNEL CALIBRATION.) This channel calibration surveillance would continue to be performed in the same manner but at a reduced frequency. No modifications to test methodologies or station equipment have been proposed in this request. This request is made to facilitate a change to the Duane Arnold Energy Center operating cycle from 18 months to 24 months. This request has been prepared following the guidance in Generic Letter 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.''

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

The proposed amendment extends the CHANNEL CALIBRATION surveillance frequency for the ADS timers from 18 months to 24 months to facilitate a change in the DAEC operating cycle from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the ADS. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change alters the frequency but not the Surveillance Requirement itself nor the way in which the surveillance is performed. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an evaluation of surveillance test results shows that the probability of exceeding the TS Allowable Value (AV) with the extended surveillance frequency is small and remains well within the setpoint methodology guideline. Therefore, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment extends the CHANNEL CALIBRATION surveillance frequency for the ADS timers from 18 months to 24 months to facilitate a change in the DAEC operating cycle from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, only the frequency will change; the Surveillance Requirement itself and the way the surveillance is performed will remain unchanged. Furthermore, a review of the maintenance history of these timers indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment extends the CHANNEL CALIBRATION surveillance frequency for the ADS timers from 18 months to 24 months to facilitate a change in the DAEC operating cycle from 18 months to 24 months. Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is considered small based on other more frequent testing, the availability of redundant systems or equipment, and the fact that there is no evidence of any existing equipment failures that would impact the availability of the ADS. Furthermore, an evaluation of surveillance test results shows that the probability of exceeding the TS AV with the extended surveillance frequency is small and remains well within the setpoint methodology guideline. Therefore, 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: Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Claudia M. Craig.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: August 30, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) safety limit minimum critical power ratio (MCPR) for two recirculation pump operation for Cycle 21.

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

The proposed Safety Limit MCPR (SLMCPR), and its use to determine the Cycle 21 thermal limits, have been derived using NRC approved methods and uncertainties. These methods do not change operation of the plant, and have no effect on the probability of an accident initiating event or transient. The basis of the SLMCPR is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR for Cycle 21 preserves the margin to transition boiling and the probability of fuel damage is not increased.

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

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

The proposed change results only from different inputs, for the Cycle 21 core reload. These methods and uncertainties have been reviewed and approved by the NRC, and do not involve any new or unapproved methods for operating the facility. No new initiating events or transients result from these changes.

The SLMCPR remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. A change in SLMCPR cannot create the possibility of any new type of accident. SLMCPR values for the new fuel cycle are calculated using previously transmitted methodology.

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

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

The margin of safety as defined in the TS bases will remain the same. The new SLMCPR was derived using NRC approved methods and uncertainties which are in accordance with the current fuel design and licensing criteria. The SLMCPR remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity.

Fuel licensing acceptance criteria for SLMCPR calculations apply to Monticello Cycle 21 in the same manner as previously applied. SLMCPRs prepared using methodology previously transmitted to the NRC ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving fuel cladding integrity. The operating MCPR limit is set appropriately above the safety limit value to ensure adequate margin when the cycle specific transients are evaluated.

Therefore, the proposed TS change does 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: Jay E. Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Claudia M. Craig.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: June 8, 2001.

Description of amendment request: The proposed amendment would delete the ''High Pressure Coolant Injection (HPCI) System Suppression Pool Water Level—High'' function from Technical Specification (TS) 3.3.5.1, ''Emergency Core Cooling System (ECCS) Instrumentation.'' This change would eliminate automatic transfer of the HPCI pump suction source from the condensate storage tank (CST) to the suppression pool for a high suppression pool level. Elimination of this function is expected to increase the availability of the HPCI system during a postulated anticipated transient without scram (ATWS) with standby liquid control system (SLCS) failure and to reduce operator burden during a postulated station blackout (SBO) event.

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 of occurrence or consequences of an accident previously evaluated?

Deletion of the automatic HPCI suction transfer from the CST to the suppression pool for a high suppression pool level condition was analyzed for impacts against all previously evaluated accidents and transients. Eliminating the automatic transfer increases the availability of the HPCI system during an ATWS event and operator burden is reduced during a postulated Station Blackout (SBO). There are no adverse effects, consequences, or changes in the probability of an accident occurring as a result of this change. HPCI operation is improved and all other plant systems remain unaffected in their ability to perform their design basis functions as a result of this change.

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

Implementation of this change increases the availability of the HPCI system during a postulated ATWS with Standby Liquid Control System (SLCS) failure. The change only affects the HPCI suction source and whether the source is automatically transferred from the preferred CST to the suppression pool for a high suppression pool level. Continued HPCI operation utilizing the CST as a suction source does not create a new or different type of accident from those previously analyzed. The primary effect of this change is to the suppression pool level which has been evaluated and found to be acceptable for all relevant accidents and transients. Therefore a new or different accident is not created and all other accident analyses are unaffected by the change.

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

This change does not reduce any margin of safety. The increase in suppression pool water level does not cause containment hydrodynamic loads to exceed design limits under accident conditions. Overall, HPCI reliability is increased as it would remain operable during the ATWS with Loss of SLCS event. This increased availability of the HPCI system provides for additional defense in depth which reduces the probability of core damage.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Peter Tam, Acting.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: July 17, 2001.

Description of amendment request: The proposed amendment would revise the reactor pressure vessel (RPV) pressure-temperature (P–T) limits specified in the technical specifications (TSs). Editorial changes associated with the P–T limit revisions are also proposed. The proposed P–T limits rely on the methodology for determining allowable P–T limits specified in American Society of Mechanical Engineers (ASME) Code Case N–640. The revised P–T limits will allow required RPV hydrostatic and leak tests to be performed at a significantly lower temperature. This is expected to reduce challenges to plant operators associated with maintaining the reactor coolant system within a narrow temperature band during testing.

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 change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The changes to the calculational methodology for the pressure and temperature (P–T) limits based upon Code Case N–640 continue to provide adequate margin in the prevention of a brittle-type fracture of the reactor pressure vessel (RPV). The code case was developed based upon the knowledge gained through years of industry experience. P–T curves developed using the allowances of Code Case N–640 indeed yield more operating margin. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the existing assumptions used for the calculation of P–T limits are unnecessarily conservative and unrealistic. Therefore, providing the allowances of the subject Code Case in developing the P–T limit curves will continue to provide adequate protection against nonductile-type fractures of the RPV.

The evaluation for the Unit 1 and Unit 2 P–T limit curves for 32 EFPYs was performed using the approved methodologies of 10 CFR 50, Appendix G. The curves generated from these methods ensure the P–T limits will not be exceeded during any phase of reactor operation. Resolution of the current industry issues related to fluence calculation methodology requires PPL to limit applicability of the curves to May 1, 2005 for Unit 2 and May 1, 2006 for Unit 1. Therefore, the probability of occurrence and the consequences of a previously analyzed event are not significantly increased. Finally, the proposed changes will not affect any other system or piece of equipment designed for the prevention or mitigation of previously analyzed events. Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes.

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

The proposed changes provide more operating margin in the P–T limit curves for inservice leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with the benefits being primarily realizable during the pressure tests. Operation in the ''new'' regions of the newly developed P–T curves has been analyzed in accordance with the provisions of ASME Code, Section XI, Appendix G; 10 CFR 50 Appendix G, and ASME Code Case N–640, thus providing adequate protection against a nonductile-type fracture of the RPV. These proposed changes do not create the possibility of any new or different type of accident. Further, they do not result in any new or unanalyzed operation of any system or piece of equipment important to safety.

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

As mentioned previously, the revised P–T curves provide more operating margin and thus, more operational flexibility than the current P–T curves. However, the industry experience since the inception of the P–T limits in 1974 confirms that some of the existing methodologies used to develop P–T curves is unrealistic and unnecessarily conservative. Accordingly, ASME Code Case N–640 takes advantage of the acquired knowledge by establishing more realistic methodologies for the development of P–T curves.

Use of Code Case N–640 to develop the revised P–T curves utilized the K_{IC} fracture toughness curve in lieu of the K_{IA} curve as the lower bound for fracture toughness. Use of the K_{IC} curve to determine lower bound fracture toughness is more technically correct than using the K_{IA} curve. P–T curves based on the K_{IC} fracture toughness limits enhance overall plant safety by expanding the P–T window in the low-temperature operating region. The benefits which occur are a reduction in the duration of the pressure test and personnel safety while conducting inspections in primary containment with no decrease to the margin of safety.

Therefore, operational flexibility is gained without a reduction in the margin of safety to RPV brittle fracture.

The development of the P–T curves to 32 EFPYs was performed per the guidelines of 10 CFR [part] 50, and thus, the margin of safety is not reduced as the result of the proposed changes. Resolution of the current industry issues related to fluence calculation methodology requires PPL to limit applicability of the curves to May 1, 2005 for Unit 2 and May 1, 2006 for Unit 1.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, Inc., 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Peter Tam, Acting.

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 requests: August 24, 2001.

Description of amendment requests: The licensee requests an amendment to the Technical Specifications for containment leakage rate testing as a result of recalculation of peak containment internal pressure following certain design-basis accidents. The purpose of the change is to make the Technical Specifications appropriately reflect up-to-date calculated peak containment pressure. The revised calculated peak containment pressure related to the design basis loss-of-coolant accident and the revised calculated peak containment pressure for the design basis Main Steam Line Break would be lower than the current Technical Specification values.

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

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

Response: No.

The proposed change would revise the Operating Licenses for San Onofre Nuclear Generating Station Units 2 and 3 to amend Technical Specification (TS) 5.5.2.15, "Containment Leakage Rate Testing Program," by changing the stated calculated values for peak containment internal pressure for the design basis Loss Of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) accident. The current LOCA value of 55.1 psig would be changed to 45.9 psig and the current MSLB value of 56.6 psig would be changed to 56.5 psig.

The proposed change does not affect the probability of occurrence of an accident previously evaluated because it relates solely to the consequences of hypothesized accidents given that the accident has already occurred.

The proposed change does not increase the calculated peak containment internal pressure for the LOCA and MSLB accidents, and thus does not increase their consequences.

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

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

Response: No.

The proposed change relates to two accidents, MSLB and LOCA, already evaluated in the Updated Final Safety Analysis Report. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The recalculated peak containment internal pressures for the MSLB and LOCA accidents are less than the containment design pressure and less than the previously calculated pressures. 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 requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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 16, 2001.

Description of amendment request: The proposed amendments delete requirements from the Technical Specifications (TS) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," and Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated August 16, 2001.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

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

The NRC staff proposes to determine that the amendment requests involve 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.

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: August 2, 2001.

Description of amendment request: The proposed amendments delete requirements from the Technical Specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated August 2, 2001.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post–TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

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

The NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

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.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and 2), Beaver County, Pennsylvania

Date of amendment request: January 18, 2001, as supplemented by letter dated June 26, 2001.

Brief description of amendment request: The proposed amendment would revise the applicability of the current BVPS–1 heatup/cooldown curves from 15 effective full-power years (EFPY) to 14 EFPY. Proposed changes to Technical Specification (TS) 3.7.1.1, ‘‘Main Steam Safety Valves (MSSVs),’’ include revisions of the limiting condition for operation and to the title and content of Table 3.7–1 to provide consistency with the improved standard TSs, creation of new Actions to address inoperable MSSVs, reduction of the power range neutron flux-high reactor trip setpoint to be consistent with TS Traveler Form—235, Revision 1, and changes to the maximum power levels permissible with inoperable MSSVs. TS Bases changes are also proposed for consistency.

Date of publication of individual notice in **Federal Register:** July 27, 2001 (66 FR 39212).

Expiration date of individual notice: August 27, 2001.

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov*.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: July 2, 2001.

Brief description of amendments: The amendments deleted Technical Specifications Section 5.5.4, ''Post Accident Sampling,'' for Catawba Nuclear Station, Units 1 and 2, and thereby eliminated the requirements to have and maintain the post-accident sampling systems.

Date of issuance: September 11, 2001.

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

Amendment Nos.: 193 and 185.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 8, 2001 (66 FR 41615).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 11, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 2, 2001.

Brief description of amendments: The amendments deleted Technical Specifications (TS) Section 5.5.4, ''Post Accident Sampling,'' for McGuire Nuclear Station, Units 1 and 2, and thereby eliminated the requirements to have and maintain the post-accident sampling systems (PASS). The amendments also delete PASS-related License Conditions 2.C(11)c, ''Post Accident Sampling (II.B.3),'' for Unit 1 and 2.C(10)b, ''Postaccident Sampling (II.B.3),'' for Unit 2.

Date of issuance: September 17, 2001.

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

Amendment Nos.: 199 and 180.

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

Date of initial notice in **Federal Register:** August 8, 2001 (66 FR 41616).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 17, 2001.

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: February 28, 2001, supplemented June 27, 2001.

Brief description of amendments: The amendments add new Technical Specification 3.3.28 and Bases B 3.3.28 governing the addition of the low pressure service water standby pump automatic start circuitry.

Date of Issuance: September 6, 2001.

Effective date: As of the date of issuance and shall be implemented before the end of the Oconee Unit 3 End of Cycle 19 Refueling Outage.

Amendment Nos.: 319, 319, and 319.

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

Date of initial notice in **Federal Register:** March 21, 2001 (66 FR 1517).

The supplement dated June 27, 2001, provided clarifying information that did not change the scope of the February 28, 2001, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 6, 2001.

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: January 24, 2001, as supplemented by letter dated March 22, 2001.

Brief description of amendment: The amendment changes the limit on the Low Power Setpoint, from 20 percent to 10 percent power, as specified in Technical Specification (TS) 3.1.3, ''Control Rod OPERABILITY,'' TS 3.1.6 ''Control Rod Pattern,'' and TS 3.3.2.1, ''Control Rod Block Instrumentation.''

Date of issuance: September 7, 2001.

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

Amendment No.: 118.

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

Date of initial notice in **Federal Register:** March 21, 2001 (66 FR 15921).

The supplemental letter dated March 22, 2001, provided additional information that did not expand the scope of the application or change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 7, 2001.

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: January 24, 2001, as supplemented by letters dated July 20 and August 7, 2001.

Brief description of amendment: The amendment request proposes changes to the Technical Specifications (TSs) concerning certain operational conditions required when conducting core alterations or handling irradiated fuel in the primary containment. In addition, the licensee proposes to implement administrative controls in accordance with draft NUMARC 93–01, ''Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,'' Revision 3, Section 11.3.6.5, ''Containment—Primary (PWR [pressurized-water reactor])/ Secondary (BWR [boiling-water reactor]),'' Revision 3, Section 11.3.6, ''Assessment Methods for Shutdown Conditions,'' in lieu of License Condition 2.C.(17) and change terms to make them consistent with the terminology in other revised TSs.

Date of issuance: September 14, 2001.

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

Amendment No.: 119.

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

Date of initial notice in **Federal Register:** April 18, 2001 (66 FR 20001).

The supplemental letters dated July 20 and August 7, 2001, provided additional information that did not expand the scope of the application or change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 2001.

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: January 24, 2001, as supplemented by letters dated July 2, and August 6 and 20, 2001.

Brief description of amendment: The license amendment request consists of changes to the Technical Specifications (TSs) to revise the reactor vessel pressure/temperature (P/T or P–T) limits specified in TS 3.4.11, ''RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits,'' for reactor heat-up. The current RCS P/T Limits in TS Figure 3.4–11, ''Minimum Temperature Required Vs. RCS Pressure,'' would be replaced with recalculated RCS P/T limits based, in part, on an alternate methodology. The alternate methodology uses American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Code) Case N–640, ''Alternative Requirement Fracture Toughness for Development of P–T Limit Curves for ASME B&PV Code Section XI, Division 1,'' for alternate reference fracture toughness for reactor vessel materials in determining the P/T limits.

Date of issuance: September 14, 2001.

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

Amendment No.: 120.

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

Date of initial notice in **Federal Register:** March 21, 2001 (66 FR 15920).

The supplemental letters dated July 2, and August 6 and 20, 2001, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: April 27, 2001.

Brief description of amendments: The amendments revise Technical Specifications (TS) Section 5.5.7, to provide an exception to the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, which would allow either a qualified in-place ultrasonic volumetric examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or liquid penetrant testing) of exposed surfaces of the removed flywheel to be conducted at approximately 10-year intervals. The proposed change is in accordance with the Nuclear Regulatory Commission (NRC) approved Improved Standard TS Generic Change Traveler TSTF–237, Revision 1.

Date of issuance: September 20, 2001.

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

Amendment Nos.: 123, 123, 118, and 118.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 20, 2001.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: June 26, 2001.

Brief description of amendments: To extend the dates specified in Operating License Sections 2.C(8) and 3.P, ''Pressure—Temperature Limit Curves,'' for Dresden Nuclear Power Station, Units 2 and 3, respectively.

Date of issuance: September 10, 2001.

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

Amendment Nos.: 187 and 182.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Facility Operating License.

Date of initial notice in **Federal Register:** The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 10, 2001.

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: March 21, 2001, as supplemented June 28, 2001.

Brief description of amendment: The amendment revised the Improved Technical Specifications (ITS) 5.6.2.10, ''OTSG [Once-Through Steam Generator] Tube Surveillance Program'' to implement a reroll process to repair degraded steam generator tubes and allow the reroll repairs to be used in both the upper and lower tubesheets.

Date of issuance: September 10, 2001.

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

Amendment No.: 198.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 18, 2001 (66 FR 20006). The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 10, 2001.

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: October 3, 2000, as supplemented June 14, August 28, and September 7, 2001.

Brief description of amendment: The amendment revised the Improved Technical Specifications (ITS) 3.7.12, ''Control Room Emergency Ventilation System (CREVS)''; ITS 5.6.2.12, ''Ventilation Filter Testing Program (VFTP)''; ITS 3.3.16, ''Control Room Isolation—High Radiation''; and ITS 3.7.18, ''Control Complex Cooling System.'' The proposed ITS changes are based on the results of revised public and control room dose calculations for CR–3 design basis radiological accidents using an alternative source term and the adoption of Technical Task Force Traveler (TSTF) 287. A new Section 5.6.2.21, ''Control Complex Habitability Envelope Program,'' is added.

Date of issuance: September 17, 2001.

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

Amendment No.: 199.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 15, 2000 (65 FR 69060). The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 17, 2001.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: April 18, 2001, as supplemented August 24, 2001.

Brief description of amendment: This amendment revises Technical Specification (TS) 6.9.1.11, ''Core Operating Limits Report (COLR),'' to include the ABB-Combustion Engineering Topical Report CENPD–387–P–A, Rev 000, in the list of analytical methods. This allows use of an improved heat flux correlation (designated ABB–NV) previously approved by the NRC. Additionally, the Bases for TS 2.1.1, ''Reactor Core,'' are modified to reflect use of the improved heat flux correlation.

Date of Issuance: September 20, 2001.

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

Amendment No.: 118.

Facility Operating License No. NPF–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 30, 2001 (66 FR 29358). The August 24, 2001, supplement did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 20, 2001.

No significant hazards consideration comments received: No.

GPU Nuclear Inc., Docket No. 50–320, Three Mile Island Nuclear Generating Station, Unit 2, Dauphin County, Pennsylvania

Date of amendment request: July 25, 2000, as supplemented by letter dated June 21, 2001.

Brief description of amendment request: The amendment changes Three Mile Island Nuclear Generating Station, Unit 2 (TMI–2), Technical Specification (TS) 6.7.2 to eliminate a change associated with periodic reviews of procedures. Currently, TS 6.7.2 states that required procedures shall be reviewed periodically as required by American National Standards Institute (ANSI) Standard N18.7–1976 (a biennial review). This amendment revises the wording for TS 6.7.2 to state that required procedures shall be reviewed periodically. This amendment is also consistent with the TMI–2 Post-Defueling Monitored Storage Quality Assurance Plan, which states that ''Procedural documentation shall be periodically reviewed for adequacy as set forth in administrative procedures.''

Date of Issuance: September 7, 2001.

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

Amendment No.: 56.

Facility Operating License No. DPR–73: Amendment revises the Technical Specification.

Date of initial notice in **Federal Register:** December 13, 2000 (66 FR 77920).

The June 21, 2001, supplemental letter replaced in its entirety the original application dated July 25, 2000. The supplement did not expand the scope of the original request.

The Commission's related evaluation is contained in a safety evaluation dated September 7, 2001.

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: August 17, 2001.

Brief description of amendment: The amendment allowed a one-time exception to Technical Specification Surveillance Requirement (SR) 3.6.1.7.2 for suppression chamber-to-drywell vacuum breakers 2ISC*RV35A and 2ISC*RV35B. A note has been added to SR 3.6.1.7.2 stating that function testing of these vacuum breakers is not required to be met for the remainder of Cycle 8.

Date of issuance: September 7, 2001.

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

Amendment No.: 98.

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

Public Comments Requested as to Proposed No Significant Hazards Consideration: Yes (66 FR 44653) August 24, 2001. That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. Comments were received from one person, and were addressed in the safety evaluation associated with the amendment. The notice also provided for an opportunity to request a hearing by September 24, 2001, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after the issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, final determination of no significant hazards consideration determination, and state consultation, are contained in a safety evaluation dated September 7, 2001.

Attorney for the Licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Peter S. Tam, Acting.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: April 6, 2001.

Brief description of amendment: The amendment changes Technical Specifications Section 5.5.10, ''Technical Specifications (TS) Bases Control Program,'' in accordance with Nuclear Energy Institute TS Task Force Standard TS Change Traveler, TSTF–364, ''Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59,'' Revision 0.

Date of issuance: September 13, 2001.

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

Amendment No.: 241.

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

Date of initial notice in **Federal Register:** August 8, 2001 (66 FR 41623).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: May 25, 2001, as supplemented August 17, 2001.

Brief description of amendment: The amendment to the Kewaunee Nuclear Power Plant Technical Specifications (TSs) 4.2 revises TS 4.2 to revise the surveillance requirements and bases for TS 4.2.b, ''Steam Generator Tubes,'' to account for changes associated with replacement of the original steam generators. Specifically, the changes delete inspection requirements associated with steam generator tube sleeving and repair limits and revise the phrasing of text within the TS to enhance clarity.

Date of issuance: September 20, 2001.

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

Amendment No.: 158.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 12, 2001 (66 FR 31711). The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 20, 2001.

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: May 24, 2001.

Brief description of amendments: The amendments revise the Technical Specification definition of CORE ALTERATIONS.

Date of issuance: September 11, 2001.

Effective date: The amendments are effective as of the date of their issuance and shall be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1—131; Unit 2—120.

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

Date of initial notice in **Federal Register:** July 11, 2001 (66 FR 36345)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 11, 2001.

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 amendments request: May 24, 2001.

Brief description of amendments: The amendments relocate Technical Specification 3/4.9.6, ''Refueling Machine'' and its associated Bases description to the Technical Requirements Manual.

Date of issuance: September 13, 2001.

Effective date: The amendments are effective as of the date of their issuance.

Amendment Nos.: Unit 1—132; Unit 2—121.

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

Date of initial notice in **Federal Register:** July 11, 2001 (66 FR 36344).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 13, 2001.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–296, Browns Ferry Nuclear Plant, Units 3 Limestone County, Alabama

Date of application for amendment: July 25, 2001.

Brief description of amendment: The proposed amendment deletes Technical Specification (TS)-required Action 3.3.1.1.I.2, which limits plant operation to 120 days in the event of the inoperability of the Oscillation Power Range Monitor trip system. For this situation, the proposed change would allow plant operation to continue if the existing TS Required Action 3.3.1.1.I.1, to implement an alternate means to detect and suppress thermal hydraulic instability oscillations, was taken.

Date of issuance: September 13, 2001.

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

Amendment No.: 231.

Facility Operating License No. DPR–68: Amendment revises the TS.

Date of initial notice in **Federal Register:** August 8, 2001 (66 FR 41627).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: June 16, 2000, as supplemented September 27, 2000, and June 6, 2001.

Brief Description of amendments: These amendments change the reactor protection system and engineered safety features actuation system analog instrumentation surveillance frequency from monthly to quarterly.

Date of issuance: August 31, 2001.

Effective date: August 31, 2001.

Amendment Nos.: 228 and 228.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in **Federal Register:** November 15, 2000 (65 FR 69067). The September 27, 2000, and June 6, 2001, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 31, 2001.

No significant hazards consideration comments received: No.

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

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

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal**

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by Email to *pdr@nrc.gov.*

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By November 2, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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 20555–001, 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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Oswego County, New York

Date of amendment request: September 14, 2001.

Brief description of amendment: The amendment authorizes a one-time-only change to Technical Specifications Section 3.9.B.1 and associated Bases. Specifically, this change extends the Limiting Condition for Operation allowable out-of-service time for one incoming Reserve AC Power line (115KV line #3) and/or one reserve station transformer inoperable from 7 days to 14 days during the period commencing September 9, 2001 and extending through September 23, 2001.

Date of issuance: September 15, 2001.

Effective date: As of the date of issuance.

Amendment No.: 272.

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

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

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration, are contained in a Safety Evaluation dated September 14, 2001.

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Peter Tam (Acting)

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

Dated at Rockville, Maryland, this 25th day of September 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–24580 Filed 10–2–01; 8:45 am]

BILLING CODE 7590–01–P

POSTAL RATE COMMISSION

[Order No. 1324; Docket No. R2001–1]

Postal Rate and Fee Changes

AGENCY: Postal Rate Commission.

ACTION: Notice and order in omnibus rate and classification case.

DATES: Notices of intervention, answers to motions, and comments on request for expedition due October 24, 2001; prehearing conference on October 25, 2001; comments regarding pending cases due October 29, 2001. See **SUPPLEMENTARY INFORMATION** section for other dates.

ADDRESSES: Send notices of intervention or comments to the Commission in care of the Acting Secretary, 1333 H Street NW., suite 300, Washington, DC 20268–0001.

SUMMARY: This document informs the public that the Postal Service has filed a request for an expedited decision on omnibus rate, fee and classification changes. It identifies overall percentage increases for various classes, encourages interested persons to review the filing to determine its impact for further details, and takes several preliminary procedural steps. It also states that a companion document will contain specific proposed rate and fee changes.

FOR FURTHER INFORMATION CONTACT: Stephen L. Sharfman, General Counsel, 202–789–6820.

SUPPLEMENTARY INFORMATION:

I. Introduction

This notice and order [no. 1323, issued September 26, 2001] informs the public that on September 24, 2001, the United States Postal Service filed a request with the Postal Rate Commission for an expedited recommended decision on proposed changes in essentially all domestic postage rates and fees, and in some mail classifications.[1] It summarizes basic features of the filing, including several contemporaneous notices and motions; institutes a formal proceeding for consideration of the Service's proposals; sets October 25, 2001 as the date for a prehearing conference; and takes several other initial procedural steps. A companion notice and order presents a complete schedule of the Service's proposed rate and fee changes.

Summary. The request affects virtually all of the Service's offerings, and is based on important assumptions regarding costs, volumes, pricing and, in some instances, classification changes. It includes a proposed 3-cent increase in the First-Class stamp, raising the price from 34 cents to 37 cents. The charge for each additional ounce of single-piece First-Class Mail would remain at 23 cents.

The Postal Service has indicated the proposed systemwide average increase for all classes of mail and services is 8.7 percent. Average increases, by individual class of mail, are 8.2 percent for First-Class Mail; 9.7 percent for Express Mail; 13.5 percent for Priority Mail; 10.0 percent for Periodicals; 7.3 percent Standard Mail; and 8.9 percent for Package Services. Proposed percentage changes for the Special Services vary considerably by individual service.

Rate changes for a specific piece of mail, bulk mailings, or a special service may differ significantly from the systemwide average change, as well as from the referenced change for an individual class of mail. Many subclasses and services include numerous individual rate cells, and the application of various discounts, surcharges, and annual mailing permit fees often determines effective percentage changes. Interested persons are urged to carefully review the Service's filing to determine the proposal's impact.

II. Establishment of Formal Docket

The Service's request was filed pursuant to sections 3622 and 3623 of the Postal Reorganization Act (39 U.S.C. 3622, 3623). The Commission hereby institutes a proceeding, designated as docket no. R2001–1, postal rate and fee changes, to consider the instant request. In the course of this proceeding, participants may propose alternatives to

[1] Request of the United States Postal Service for a recommended decision on changes in rates of postage and fees for postal services, September 24, 2001 (Service's request or request).