Reconsideration of a May 3, 1999 Order'' (Public Meeting) be held on May 27, and on less than one week's notice to the public.

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The NRC Commission Meeting Schedule can be found on the Internet at: http://www.ncr.gov/SECY/smj/schedule.htm

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, DC 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: May 27, 1999.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 99–14041 Filed 5–28–99; 12:10 pm]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 8, 1999, through May 20, 1999. The last biweekly notice was published on May 19, 1999.

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 Administration Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By July 2, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: May 5, 1999.

Description of amendment request: The proposed amendment would change the technical specifications (TS) and licensing basis for the required amount of diesel fuel to be stored on-site and its sources.

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

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

The proposed change affects only the on-site diesel fuel storage capacity for the operation of emergency diesel generators [EDG]. The on-site storage capacity is not associated with an accident precursor/initiator; thus, it has no impact on the probability of [an] accident occurring. The consequences of an accident would not be significantly increased because reasonable measures will be available to ensure the EDGs are supplied with enough fuel from the on-site sources to operate for seven days at rated capacity.

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

The proposed change does not affect normal plant operation or the immediate response to an accident. The only change is the proposed refilling operation to transfer fuel from the Class II SBODG [Station Blackout Diesel Generators] storage tanks to the Class I EDG tanks. The refilling operation would occur entirely outdoors through above ground hoses connecting the EDG and SBODG tanks. This operation would only be required following a LOCA [loss-of-coolant accident], an accident already analyzed. Since the proposed refilling operation is a post-accident evolution, it would not be in place to cause an accident of a different type during non-accident conditions. No reasonable malfunction of equipment associated with the evolution could create a new or different kind of accident than previously evaluated.

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

The proposed amendment for licensing basis change and TS change does not significantly reduce the margin of safety. The proposed change restores the licensing basis to provide sufficient fuel in on-site storage tanks for continuous operation of each EDG for approximately seven days. The revised licensing basis requires 36,800 gallons of fuel per EDG to be stored on-site. A minimum of 19,800 gallons of fuel will be stored in Class I EDG storage tanks and the remaining will be stored in Class II SBODG on-site storage tanks. The storage of fuel in Class I tanks does not reduce the margin of safety. The only potential reduction in the margin of safety is due to the use of Class II SBODG tanks and associated transfer equipment for the storage and transfer of additional fuel. These Class II tanks are rugged, double-wall fiberglass tanks. While not designed to safety-related requirements, the failure of these tanks under extreme environmental conditions, such as an earthquake, has been evaluated to be very unlikely. Thus, on-site storage of sufficient fuel for operation of both EDGs is assured to mitigate the consequences of an accident previously evaluated. All stored fuel is maintained at the same quality standard. The proposed diesel fuel refilling operation is a post design basis accident activity, which does not create the possibility of a new accident or impact an accident previously evaluated. Therefore, there is no significant reduction in the safety margin.

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

Local Public Document Room location: Plymouth Public Library, 132 South Street, Plymouth, Massachusetts 02360.

Attorney for licensee: J. Fulton, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Section Chief: James W. Clifford.

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of application of amendment request: October 2, 1998, as supplemented by letter dated April 19, 1999.

Description of amendment request: By letter dated February 13, 1998, Commonwealth Edison Company (ComEd) certified that they have permanently ceased operations at Zion Nuclear Power Station (ZNPS), Units 1 and 2. Since ComEd has permanently ceased operations at ZNPS, they have requested an amendment to the Facility Operating Licenses to eliminate license conditions that are no longer applicable and to replace the existing technical specifications in their entirety with permanently defueled technical specifications (PDTS). The PDTS reflect the permanently shutdown and defueled condition of the ZNPS.

Basis for a 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 and has determined that the proposed changes do not:

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

The administrative changes remove requirements that are not invoked with the reactors permanently defueled. The editorial changes alter format, word choice, grammar, terminology, etc., but do not change requirements. The more restrictive changes add new requirements, remove existing exceptions, or make existing limits more conservative. The relocation or redundancy changes remove requirements from the facility operating licenses or technical specifications because they exist in another document controlled by other approved methods. None of these types of changes affect the probability or consequences of a previously evaluated accident since there is no functional reduction in the limitations imposed on structures, systems, components or activities with the reactors permanently defueled.

The less restrictive changes to the license conditions eliminate requirements for programs and commitments that address hazards or conditions that are no longer credible with both reactors permanently defueled. Since these hazards or conditions are not credible, no increase in the probability or consequences of a previously evaluated accident will result from the elimination of these requirements.

The less restrictive changes to the equipment-related technical specifications eliminate or modify restrictions involving certain structures systems and components (SSCs). Some of the equipment-related technical specifications have been eliminated because, with both reactors permanently defueled, the spectrum of previously evaluated credible accidents has been significantly reduced and many of the associated hazards (such as reactor coolant gaseous activity, hydrogen, and radioactive iodine) will not occur. Since those previously evaluated accidents and associated hazards are no longer credible, their probability and consequences are not increased by the changes eliminating the associated technical specifications. Other equipment-related technical specifications have been modified to address previously evaluated accidents that are still relevant in the permanently defueled condition more logically and consistently, without increasing their probability or consequences.

The less restrictive changes to the Administrative Control technical specifications affect a variety of functions. They provide flexibility in Quality Assurance Program administration, allow a reduction in shift staffing, eliminate certain training requirements for personnel who have little or no safety involvement, change certain procedure processing requirements, provide consistency in scheduling certain radiological surveillances and reports, eliminate reports that are no longer needed, eliminate unnecessary flood door requirements, and allow alternative methods of administering Process Control Program changes. Since none of these changes directly involve the previously evaluated accidents that remain credible with both reactors permanently defueled, the changes will not increase the probability or consequences of any previously evaluated accident.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The administrative changes do not alter any SSCs or activities involved with the safe storage of nuclear fuel. The editorial changes do not alter any requirements. The more restrictive changes make the technical specifications more limiting. The relocation/redundancy changes only change the location of requirements. None of these types of changes create the possibility of a new or different kind of accident from any accident previously evaluated.

The less restrictive changes to the license conditions eliminate requirements for programs and commitments involving hazards or conditions that are no longer credible with both reactors permanently defueled. Since these changes do not result in any new programs or activities, they do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The less restrictive changes to the equipment-related technical specifications do not alter any SSC or cause any SSC to be operated in a manner that could initiate any event or accident. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The less restrictive changes to the Administrative Control technical specifications do not change the design, function, or operation of any SSC except the flood doors and the change involving the flood doors does not introduce any new type of event. Therefore, the less restrictive changes to the Administrative Control technical specifications do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The administrative changes do not alter any SSCs or activities involved with the safe storage of nuclear fuel. The editorial changes do not alter any requirements. The more restrictive changes make the technical specifications more limiting. The relocation/redundancy changes only change the location of requirements. None of these types of changes reduce any safety margin.

The less restrictive changes to the license conditions eliminate requirements that apply to hazards or conditions that are no longer relevant with both reactors permanently defueled. The safety margins that may have been associated with those license conditions are no longer relevant.

There are no longer any relevant margins of safety associated with the less restrictive changes to the equipment-related technical specifications except for those involving criticality control and seismic criteria. The proposed technical specifications maintain the same margin of safety for criticality control in the spent fuel pool, and the Defueled Safety Analysis Report imposes seismic criteria that provide an adequate safety margin.

The less restrictive changes to the Administrative Control technical specifications do not directly involve any limits or parameters and therefore cannot affect any margin of safety.

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

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

Attorney for licensee: Pamela B. Strobel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–076.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: May 14, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specification requirements affecting the surveillance criteria for that portion of the once-through steam generator tubes regarded as a primary-to-secondary pressure boundary located within the upper tube sheet and impacted by a specific degradation mechanism, namely, outside diameter intergranular attack.

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

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

The once-through steam generators OTSG are used to remove heat from the reactor coolant system during normal operation and during accident conditions. The OTSG tubing forms a substantial portion of the reactor coolant pressure boundary. An OTSG tube failure is a breach of the reactor coolant pressure boundary and is a specific accident analyzed in the ANO–1 [Arkansas Nuclear One, Unit 1] Safety Analysis Report.

The purpose of the periodic surveillance performed on the OTSGs in accordance with ANO–1 Technical Specification (TS) 4.18 is to ensure that the structural integrity of this portion of the reactor coolant system will be maintained. The TS plugging limit of 40% of the nominal tube wall thickness requires tubes to be repaired or removed from service because the tube may become unserviceable prior to the next inspection. Unserviceable is defined in the TS as the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an operating basis earthquake, a loss-of-coolant accident, or a steam line or feedwater line break. The proposed TS change allows OTSG tubes with ODIGA [outside diameter intergranular attack] indications contained within a defined area of the UTS [upper tube sheet] to remain in service with existing degradation exceeding the existing 40% through-wall (TW) plugging limit.

Extensive testing and plant experience has illustrated that ODIGA flaws confined to this area within the OTSG will not result in tube burst or tube leakage. Therefore, allowing ODIGA flaws in this specific region to remain in service will not alter the conditions assumed in the current ANO–1 accident analysis for OTSG tube failures under postulated accident conditions. In addition, the condition of the OTSG tubes in this region are monitored during regular inspection intervals to assess for evidence of growth. Any growth noted will be addressed through testing and the operational assessment * * *.

Application of the ODIGA alternate repair criteria will allow leaving tubes with ODIGA indications found in the defined area of the UTS in service while ensuring safe operation by monitoring and assessing the present and future conditions of the tubes. ANO–1 has operated since 1984 with ODIGA affected tubes in service with no appreciable effect on structural integrity or indications of tube leakage from ODIGA sources within the UTS. Through the inspection, testing, monitoring, and assessment program previously mentioned, and the on-line leak detection capabilities available during plant operation, continued safe operation of ANO–1 is reasonably assured.

Therefore, the application of the ODIGA alternate repair criteria...does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The implementation of the ODIGA alternate repair criteria will not result in any failure mode not previously analyzed. The OTSGs are passive components. The intent of the TS surveillance requirements are being met by these proposed changes in that adequate structural and leak integrity will be maintained. Additionally, the proposed change does not introduce any new modes of plant operation.

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

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

The application of an alternate repair criteria for ODIGA provides adequate assurance with margin that ANO–1 steam generator tubes will retain their integrity under normal and accident conditions. The structural requirements of ODIGA affected tubes have been evaluated satisfactorily and meet or exceed regulatory requirements. Leakage rates for these tubes within the defined region of the upper tubesheet are essentially zero and are reasonably assured to remain within the assumptions of the accident analysis by proper application of the ODIGA alternate repair criteria program. Because no appreciable impact is evidenced on the tubes structural integrity or its resulting leak rate, the margin to safety remains effectively unaltered.

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

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

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 5, 1999.

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 (CR–3) Improved Technical Specifications to approve an alternate repair criteria (ARC) for axial tube end crack-like indications in the upper and lower tubesheets of the CR–3 Once Through Steam Generators (OTSGs). The ARC will allow leaving OTSG tubes with axially oriented tube end cracks located within the clad region of the tube-to-tubesheet roll joint in service. Tubes with crack-like indications within the carbon steel portion of the tubesheet, or tubes with circumferentially oriented tube end cracks or volumetric indications within the Inconel clad region of the tubesheet, would be repaired or removed from service.

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

This LAR [License Amendment Request] proposes to implement an alternate repair criteria (ARC) for Once Through Steam Generator (OTSG) tubes with axial tube end crack (TEC) indications. Application of the ARC will allow tubes with axially oriented TEC to remain in service in accordance with specific conditions. Based on a combination of structural analyses, mock-up testing and inservice inspections, as detailed in Topical Report BAW–2346P, allowing tubes with TEC indications to remain in service is safe and justified.

Potential leakage from tubes with TEC will be bounded by the main steam line break (MSLB) evaluation presented in the Final Safety Analysis Report (FSAR). The proposed change requires inspections during subsequent outages of tubes remaining in-service with the TEC indications. The addition of this inspection does not change any accident initiators. The proposed inspection of these indications during the subsequent OTSG inservice inspections assures continuous monitoring of these tubes such that degradation of tubes containing TEC indications will be detected. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed alternate repair criteria for axial TEC indications introduces no new failure modes or accident scenarios. Topical Report BAW–2346P demonstrated structural and leakage integrity for all normal operating and accident conditions for Crystal River Unit 3 (CR–3). Furthermore, leaving TEC in service does not change the design or operating characteristics of the OTSGs. In the unlikely event that a tube with a TEC should fail and sever completely, the tube would remain engaged in the tubesheet bore, preventing interaction with other surrounding tubes. In this case, leakage is bounded by the steam generator tube rupture (SGTR) accident analysis. Therefore, this change does not create a possibility of a new or different kind of accident from any previously evaluated.

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

The mechanical joint is constrained within the tubesheet bore; thus, there is no additional risk associated with tube rupture. ITS [Improved Technical Specifications] Bases 3.4.12 contains relevant information pertaining to limitations on Reactor Coolant System leakage. The accident leakage is shown to be less than one gallon per minute primary-to-secondary leakage. Therefore, the FSAR analyzed accident scenarios remain bounding, and the use of the proposed alternate repair criteria does not reduce 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC–A5A, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Sheri R. Peterson.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: April 26, 1999.

Description of amendment request: The proposed amendments would revise the Turkey Point Plant, Units 3 and 4, Facility Operating Licenses and the Technical Specifications (TS): (1) To remove a part of license condition 3.L that is obsolete, (2) to update the TS Index to reflect all changes made to the TS Sections, TS Figures, and TS Tables by previously approved license amendments, and (3) to remove Table and Figure numeration inconsistencies found in TS 3/4.1.2.5 and TS 3/4.7.6. These proposed changes represent an administrative update to the Turkey Point Plant, Units 3 and 4, Facility Operating Licenses and to the TS.

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

(1) 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 do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes are administrative in nature removing obsolete references in the license conditions, updating the Technical Specification (TS) Index to reflect the revisions made to the TS Sections, Tables, and Figures via previous TS amendments. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications 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 use of the modified specifications can not create the possibility of a new or different kind of accident from any previously evaluated since 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.

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

The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The proposed changes to the Facility Operating License Conditions and to the Technical Specifications are administrative and do not significantly reduce any of the margins of safety.

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

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

NRC Project Director: Herbert N. Berkow.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: March 1, 1999.

Description of amendment request: Changes are proposed to support a modification which will install a digital Power Range Neutron Monitoring (PRNM) system and incorporate long-term thermal-hydraulic stability solution hardware.

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:

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

As discussed in the LTR [licensing topical report], the [Nuclear Measurements Analysis and Control] NUMAC PRNM modification and associated changes to the TS [technical specifications] involve equipment that is designed to detect the symptoms of certain events or accidents and initiate mitigating actions. The worst case failure of the equipment involved in the modification is a failure to initiate mitigating action (scram or rod block), but no failure can cause an accident. The PRNM replacement system is designed to perform the same operations as the existing Power Range Monitor system and meets or exceeds all operational requirements. Therefore, it is concluded that the probability of an accident previously evaluated is not increased as a result of replacing the existing equipment with the PRNM equipment.

The PRNM system reduces the need for tedious operator actions during normal conditions and allows the operator to focus more on overall plant conditions. The automatic self-test and increased operator information provided with the replacement system are likely to reduce the burden during off-normal conditions as well. The replacement equipment qualifications fully envelope the environmental conditions, including electromagnetic interference, in the PBAPS [Peach Bottom Atomic Power Station] control room.

The replacement equipment has been specifically designed to assure that it fully meets the response time requirements in the worst case. As a result, due to statistical variations resulting from the sampling and update cycles, the response time is typically faster than required in order to assure that the required response time is always met.

Setpoints are changed only when justified by the improved equipment performance specifications and by setpoint calculations which show that safety margins are maintained. There is no impact to the Control Rod Drop accident analysis because the PRNM system maintains all existing system functions with a reliability equal to or better than the existing Power Range Monitor system.

The replacement equipment includes up to 5 LPRM [Local Power Range Monitor] inputs on a single module compared to one per module on the current system. Up to 17 LPRM signals are processed through one preprocessor. The recirculation flow signals are processed in the same hardware as the LPRM processing. The net effect of these architectural aspects is that there are some single failures that can cause a greater loss of ''sub-functionality'' than in the current system. Other architectural and functional aspects, however, have an offsetting effect. Redundant power supplies are used so that a single failure of AC power has no effect on the overall PRNM system functions while still resulting in a half scram as does the current system. Continuous automatic self-test also assures that if a single failure does occur, it is much more likely to be detected immediately. The net effect is that from a total system level, unavailability of the safety-related functions in the replacement system is equal to or better than the current Power Range Monitor system.

Based on the extensive and through [sic] [thorough] verification and validation program used in the PRNM design and field operating experience, common cause failures in software controlled functions are judged to not be a significant failure mode. However, in spite of that conclusion, means are provided within the system to mitigate the effects of such a failure and alert the operator. Therefore, such a failure, even if it occurred, will not increase the consequences of a previously evaluated accident.

To reduce the likelihood of common cause failure of software controlled functions, thorough and careful verification and validation (V&V) activities are performed both for the requirements and the implementing software design. In addition, the software is designed to limit the loading that external systems or equipment can place on the system, thus significantly reducing the risk that some abnormal dynamic condition external to the system can cause system functional performance problems due to processing ''overload'' (i.e., ''slowing down'' or stopping the processing).

As a conservatism, however, despite these V&V activities, common cause failures of software controlled functions due to residual software design faults are assumed to occur. Both the software and hardware are designed to manage the consequences of such failure (and also cover potential common cause hardware failures). Safety outputs are designed to be fail safe by requiring dynamic update of output modules or data signals, where failure to update the information is detected by simple receiving hardware, which, in turn, forces a trip. This aspect covers all but rather complex failures where the software or hardware executes a portion of the overall logic but fails to process some portion of new information (inputs ''freeze'') or some portion of the logic (outputs ''freeze'').

To help reduce the likelihood of complex failures, a watchdog timer is used which is updated by a very simple software routine that in turn monitors the operational cycle time of all tasks in the system. The software design is such that as long as all tasks are updated at the design rate, it is likely that software controlled functions are executing as intended. Conversely, if any task fails to update at the design rate, that is a strong indication of at least some unanticipated condition. If such a condition occurs, the watchdog timer will not be updated, the computer will be automatically restarted, and the system will detect an abnormal condition and provide an alarm and trip.

The information available to the operator is at least the same as with the current system and, in many cases, improved. No actions are required by the operator to obtain information normally used and equivalent to that available with the current equipment. However, the replacement system does provide more directly accessible information regarding the condition of the equipment, including automatic self-test, which can aid the operator in diagnosing unusual situations beyond those defined in the licensing basis.

In summary, the reliability of the new PRNM system and its ability to detect and mitigate abnormal flux transients have either remained the same or improved over the existing Power Range Monitor system. Since these postulated reactivity transients are mitigated by the new system as effectively and reliability [sic] [reliably] as the existing system, the consequences of these transients have not changed. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

PBAPS Modification P00507 uses digital processing with software (firmware) control for the main signal processing part of the modification. The remainder of the equipment in the modification uses conventional equipment similar to the current system (e.g., penetrations, cables, interface panels).

The digital equipment has ''control'' processing points and software controlled digital processing where the current system has analog and discrete component processing. The result is that the specific failures of hardware and potential software common cause failure are different from the current system. The effects of software common cause failure are mitigated by hardware design and system architecture, but are of a ''different type'' of failure than those evaluated in the PBAPS Updated Final Safety Analysis Report (UFSAR). In general, the PBAPS UFSAR assumes simplistic failure modes (relays for example) but does not specifically evaluate such effects as self-test detection and automatic trip or alarm. Therefore, the replacement system may have a malfunction of a different type from those evaluated in the PBAPS UFSAR [* * *]. However, when these PRNM failures are evaluated at the system level, there are no new effects.

PBAPS Modification P00507 involves equipment that is intended to detect the symptoms of certain transients and accidents and initiate mitigating action. The worst case failure of the equipment involved in the modification is a failure to initiate mitigating action (scram), but no failure can cause an accident. This is unchanged from the current system. Software common cause failures could cause the system to fail to perform its safety function, but this possibility is addressed in Section (i) above. In that case, it might fail to initiate action to mitigate the consequences of an accident, but would not cause one. No new system level failure modes are created with the PRNM system.

Therefore, PBAPS Modification P00507 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The PRNM system response time and operator information is either maintained or improved over the current Power Range Monitor system.

The PRNM system has improved channel trip accuracy compared to the current system and meets or exceeds system requirements assumed in setpoint analysis. The channel response time exceeds the requirements.

The channel indicated accuracy is improved over the current system and meets or exceeds all of the system requirements.

The PRNM system was developed to detect the presence of thermal-hydraulic instabilities and automatically initiate the necessary actions to suppress the oscillations prior to violating the MCPR Safety Limit. The NRC has reviewed and approved the LTR concluding that the PRNM system will provide the intended protection.

Therefore, PBAPS Modification P00507 does not result in 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.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Attorney for Licensee: J.W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: January 28, 1999, as supplemented April 29, 1999, and May 17, 1999. This notice supersedes a previous notice (64 FR 19563) published April 21, 1999, which was based upon the licensee's application for amendment dated January 28, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 Technical Specifications (TSs) proposes to reduce the number of Emergency Diesel Generators (EDGs) required to be operable during cold shutdown from 2 to 1 under certain conditions.

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

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

Response: No. The equipment, which is affected by the proposed Technical Specification change, is not an initiator to those accidents postulated to occur during Cold Shutdown or Refueling operating conditions. A comprehensive systems review and EDG loading electrical analysis has demonstrated the ability of those shutdown support systems, necessary to provide safe shutdown needs, to perform their safety functions for the postulated accidents during Cold Shutdown and Refueling conditions. One EDG can support the necessary electrical loads required in Cold Shutdown and Refueling in the event of postulated accidents along with a LOOP [loss of offsite power] in the time frame required to prevent reactor core/cavity/SFP [spent fuel pit] heatup concerns. This EDG support relies upon existing plant designed manual closure of 480VAC EDS [electrical distribution system] bus tie breakers to allow a single EDG to pick up other 480VAC EDS bus loads, such as supplying an RHR [residual heat removal] pump and SFP cooling pump, located on 480VAC EDS buses 3A, 5A, or 6A. Together, operability of the required offsite circuit(s) and one EDG along with necessary portions of the AC, DC and 120 VAC vital instrument bus electrical power distribution subsystems ensures the availability of sufficient electrical sources to operate the unit in a safe manner and to mitigate the consequences of postulated accidents during shutdown (e.g., Fuel Handling Accidents), as well as other postulated events. Action statements provide prompt, specific guidance to ensure sufficiently conservative plant response should the expected EDG power supply or required offsite power supply feeders or necessary portions of AC, DC and 120 VAC vital instrument bus electrical power distribution subsystems not be available. These Action Statements are similar to those in the STS [Standard Technical Specifications]. Therefore, the proposed license amendment (i.e., changes to 3.7.F.4 and the added sections of 3.7.F.5 & 3.7.F.6) does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

Response: No. The proposed license amendment does not involve any physical changes to plant systems or component set points. The use of 480VAC EDS bus tie breakers to power loads from necessary energized 480VAC bus(es) is part of present plant design and included within the present LOOP Off-Normal operating procedures when the reactor is in Cold Shutdown operating conditions. As discussed in the Standard Technical Specifications, NUREG 1431, during plant shutdown with one EDG, it is not required to assume a single failure and concurrent loss of all offsite or all onsite power. Worst case bounding events are deemed not credible in Cold Shutdown and Refueling conditions because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and ultimately result in minimal consequences. The lone EDG is capable of accepting and starting required loads within the assumed loading sequence intervals and in the time frame required to prevent reactor core/cavity/SFP heatup concerns, with sufficient ''kW loading''. Action statements provide prompt, specific guidance to ensure sufficiently conservative plant response should the expected EDG or offsite supply feeder or the necessary portions of the AC, DC and 120 VAC vital instrument bus electrical power distribution subsystems not be available. These action statements are similar to those in the STS. Therefore, the proposed license amendment (i.e., changes to 3.7.F.4 and added sections 3.7.F.5 & 3.7.F.6) does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No. The electrical power system specifications support the equipment required to be operable, commensurate with the current level of safety, including the equipment requiring an EDG backed power source. The design review results demonstrate that operation in the conditions of Cold Shutdown and Refueling, in accordance with the proposed Technical Specification change, is acceptable from an accident mitigation standpoint. The basic system functions in Cold Shutdown and Refueling operating conditions are not changed. One EDG, along with the necessary portions of the AC, DC and 120 VAC vital instrument electrical power distribution subsystems available, can supply the necessary electrical power requirements during these plant operating conditions, and in the time frame required to prevent reactor core/cavity/SFP heatup concerns, with sufficient ''kW loading''. The analysis conducted shows that the systems are capable of performing their design basis functions. Applicable safety analysis in the Standard Technical Specifications, NUREG 1431, discusses these system requirements as well (i.e., it is not required to assume a single failure and concurrent loss of all offsite or all onsite power). Action statements, similar to those in the Standard Technical Specifications, provide prompt, specific operator actions to ensure sufficiently conservative plant response should the expected EDG power supply or the required offsite power supply feeders or AC, DC and 120 VAC vital instrument bus electrical power distribution subsystems not be available. On this basis, the proposed license amendment (i.e., changes to 3.7.F.4 and added sections 3.7.F.5 & 3.7.F.6) 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: April 6, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to change sections 3.7.A.5 and 3.7.F.4 by removing the words ''three individual underground'' and ''underground'' from the limiting conditions for operation (LCO) when referring to the emergency diesel generator (EDG) fuel oil storage tanks (FOSTs).

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

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

No. The proposed change would not change the design configuration or function of the permanently installed EDG FOSTs. The revision of TS 3.7.A.5 and 3.7.F.4 to remove the descriptive words ''three individual underground'' and ''underground'' from the text of the two LCOs is intended as a line item change, to remove unnecessarily restrictive wording in the TS. While the Standard Technical Specifications (STS), NUREG–1431, mentions in the Bases section that ''all outside tanks, pumps, and piping are located underground'', the specification itself does not contain this requirement. The intent of this TS change is to allow for, if acceptable under 10CFR50.59, the potential installation of an alternate above ground FOST to an EDG if needed to perform repairs/testing of the permanently installed FOST. This alternate tank would need to be qualified and have the required capacity to maintain the associated EDG operable. This potential modification would include design of the temporary tank to preclude winds loads from a tornadic event causing the associated EDG to become inoperable. Installation of this temporary tank would then permit repair work or replacement of an installed EDG FOST, or subsequent similar work on either of the other EDG FOSTs, one at a time. The changes to the Bases for Specification 3.7 are consistent with the change in the LCO Specification and do not alter the design or functionality of the existing EDG FOSTs. The revised LCOs are consistent with the STS in that the FOSTs will no longer be identified as ''three individual underground''. Control of future modifications to support EDG FOST work would ensure proper licensing and design basis compliance in accordance with the change process of 10CFR50.59. The associated changes of the TS Bases provide clarification regarding the normal underground configuration of the EDG FOSTs. The proposed TS change will not reduce the ability of any system, structure, or component in preventing or mitigating a design basis accident since no plant features are being altered in conjunction with this change, and future changes would be evaluated under 10CFR50.59. The description of the FOSTs, including the fact that they are underground, remains part of the current licensing basis because it is described in FSAR [final safety analysis report] section 8.2.

Therefore, the proposed changes to the TS will not result in an increase in the probability or consequences of any previously evaluated accidents. The other changes to the TS pages are editorial only, moving text to different pages.

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

No. The proposed change would not change the design configuration or function of the permanently installed EDG FOSTs. The changes to TS 3.7 and its bases in describing the physical location of the EDG FOSTs will not alter the required design criteria of these tanks nor their ability to withstand the effects of a tornado. These changes will not reduce the ability of the EDG's in meeting their design requirements of providing emergency power towards mitigating an accident. The intent of these changes is to permit the potential use of a temporary above ground FOST(s) to supply the EDGs and to fulfill the intent and requirements of the present EDG fuel oil storage system while allowing for maintenance on an EDG FOST. The 10CFR50.59 change process will be used to determine this potential modification acceptability. The intent of the temporary configuration of an above ground FOST would be to maintain the fuel oil system and EDG operable. The associated changes to the Bases section of TS 3.7 provide additional clarification of the ''underground'' nature of the EDG FOSTs. Neither the changes to the LCO in describing the EDG FOSTs (whether the normal underground tanks or any temporary above ground FOSTs) nor any changes to the TS Bases (which do not alter the design or operation of the EDG fuel oil transfer system) will affect the ability of the EDGs to provide the necessary power for operation of equipment required for mitigating previously analyzed accident scenarios. No plant features, or FSAR description of such, are being altered in conjunction with this change, and future changes would be evaluated under 10CFR50.59. Therefore, the proposed changes will not result in an unanalyzed condition and does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

No. The proposed changes will not alter any assumptions, initial conditions, or the results of any accident analyses. The design and licensing requirements for the EDG fuel oil storage system are defined in other parts of the IP3 licensing and design basis, specifically in FSAR section 8.2. Potential modifications supported by this change would require a subsequent safety evaluation in accordance with 10 CFR 50.59 regarding the design requirements (e.g., fire loads, tornadic wind loads, tornado missile criteria, security, etc.) for an alternate FOST if repairs to present ''underground'' FOSTs are undertaken. The proper design criteria for the presently installed EDG FOSTs or for potential, alternate EDG FOSTs will be maintained via present licensing and design basis requirements and through the 10 CFR 50.59 change process as required. No plant features are being altered in conjunction with this change, and future changes would be evaluated under 10 CFR 50.59. Therefore, this proposed license amendment will not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: April 9, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 Technical Specifications (TSs) proposes to increase the allowed outage time (AOT) for any one safety injection pump from 24 hours to 72 hours.

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

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

Response: The proposed 72-hour allowed outage time for any one safety injection pump does not involve a significant increase in the probability or consequences of an accident previously analyzed. The plant Technical Specifications provides allowed outage times for systems and components to accommodate preventive or corrective maintenance. A variation in the allowed outage time is not an accident initiator and thus does not result in a significant increase in the probability of an accident previously analyzed. The proposed change provides for an increase in allowed outage time for any one safety injection pump. The operability of the remaining two safety injection pumps is required by the Technical Specifications during this period. The Indian Point 3 High Head Safety Injection System consists of three safety injection pumps, each capable of providing 50 percent of the Emergency Core Cooling System [ECCS] design flow requirement. Therefore, with only one pump inoperable the remaining two pumps are capable (assuming that no single failure occurs during the period of the allowed outage time) of mitigating the consequences of previously analyzed accidents. In addition, a 72-hour allowed outage time for safety injection pumps was evaluated by the NRC (Reference 3) [NRC Memorandum, R.L. Baer to V. Stello, ''Recommended Interim Revisions to LCOs for ECCS Components,'' dated December 1, 1975] and generically approved in the Standard Technical Specifications (Reference 1) [NUREG–1431 ''Standard Technical Specifications—Westinghouse Plants,'' Revision 1, dated April 1995].

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

Response: The proposed 72-hour allowed outage time for any one safety injection pump does not create the possibility of a new or different kind of accident from any accident previously evaluated. Changing the allowed outage time is accomplished through administrative changes, such as changes to plant procedures that implement Technical Specification requirements for allowed outage time. This change does not require physical changes to plant systems or components and also does not involve changes to plant setpoints. This change also does not affect how the safety injection pumps are operated under design basis accident conditions. Therefore there are no changes resulting from the proposed new allowed outage time that alter system operation or that could create the possibility of a new or different kind of accident. In addition, a 72-hour allowed outage time safety injection [pump] was generically approved in the Standard Technical Specifications.

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

Response: The proposed 72-hour allowed outage time for any one safety injection pump does not involve a significant reduction in a margin of safety. With one safety injection pump inoperable, the remaining two pumps are capable of providing 100% of the fuel cooling flow assumed for pertinent accident analyses with the provision that the single-failure assumption is relaxed during the time period of the allowed outage time. The acceptability of a 72-hour allowed outage time for ECCS components was established in an NRC reliability analysis (Reference 3) [NRC Memorandum, R.L. Baer to V. Stello, ''Recommended Interim Revisions to LCOs for ECCS Components,'' dated December 1, 1975]. The use of the 72-hour allowed outage time was generically approved in the Standard Technical Specifications.

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

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: April 14, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3/4.9.12, ''Fuel Handling Area Ventilation System (FHAVS),'' to (1) reflect the latest filter testing standards in the test requirements, (2) add, modify, or delete certain surveillance test requirements, and (3) clarify the information in the applicable TS Bases section. The proposed amendments would also make the TS requirements more consistent with the system design basis.

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

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

A Fuel Handling Accident, as described in the Updated Final Safety Analysis Report (UFSAR) Section 15.4.6, is the design basis accident considered for establishing system configuration and performance capability for the FHAVS. This accident is defined as the dropping of a spent fuel assembly onto the spent fuel rack resulting in a rupture of the cladding of all the spent fuel rods in the assembly.

The probability of a fuel handling accident is independent of the changes proposed in this submittal and it is unaffected by this submittal. The consequences of a dropped fuel rod are significantly reduced by pre-aligning the system to its design basis function prior to moving fuel in the fuel handling building. Pre-aligning the system eliminates the potential detrimental consequences associated with a single failure of an active component on the filter train. The proposed change will not change the way the FHAVS functions to control the release of radioactive gaseous effluents. Filter testing is improved by applying more current filter testing requirements to both Units 1 and 2.

The proposed change will not modify equipment used to store or move irradiated fuel assemblies, or equipment used to move heavy loads in the Fuel Handling Building. The proposed new surveillance will be incorporated into a new or existing procedure.

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

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

The proposed change does not result in any design or physical configuration changes to the FHAVS, or to the equipment used to store or move irradiated fuel within the Fuel Handling Building. Pre-aligning the system to its design basis function prior to moving fuel in the fuel handling building eliminates the potential detrimental consequences associated with a single failure of an active component on the filter train. The system will not be operated or placed in a configuration that is different from the configuration that it was designed to operate.

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

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

The proposed changes will ensure that the FHAVS is operated and tested in accordance to its design basis requirements as specified in the Salem UFSAR.

The proposed changes will clarify the requirements of the system to be considered operable to ensure that the FHAVS will perform its intended safety function in the event of a Fuel Handling Accident. These changes ensure that the existing margin is maintained and improved by pre-aligning the system to its accident configuration.

The proposed change does not involve the addition or modification of plant equipment. It is consistent with the intent of the existing TS, the design basis of the FHAVS as described in the UFSAR, and the [Standard Technical Specifications Westinghouse Plants, NUREG–1431] ITS and associated Bases.

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

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

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, Somervell County, Texas

Date of amendment request: May 4, 1999.

Brief description of amendments: The proposed license amendments would change the CPSES Units 1 and 2 Technical Specifications. The first change revises Surveillance Requirement (SR) 3.8.4.7 to allow the unrestricted substitution of the modified battery performance discharge test in lieu of the service discharge test. The second change revises SRs 3.8.1.7, 3.8.1.12 , 3.8.1.15, and 3.8.1.20 to separate the voltage and frequency acceptance criteria for the Diesel Generator (DG) start surveillances into two sets of criteria; those criteria required to be met within 10 seconds, and those criteria required to be met following achievement of steady state conditions. The third change corrects miscellaneous editorial errors resulting from issuance of Amendment No. 64.

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 consequence of an accident previously evaluated?

(1) Batteries are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the batteries. The allowance to perform the modified performance discharge test in lieu of the service test at any time is permissible since the test's discharge rate envelopes the duty cycle of the service test. Therefore, the allowance for unrestricted substitution of the modified performance discharge test in lieu of the service discharge test does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) The diesel generators are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change does not affect the accident analysis assumption that the DG reaches minimum conditions to accept load within 10 seconds. The ability of the DG to maintain steady state operation within 10 seconds is not an accident analysis assumption and is primarily used to identify degradation of governor and voltage regulator performance. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(3) The editorial changes are non-technical and therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

(1) The allowance for unrestricted substitution of the modified performance discharge test in lieu of the service discharge test does not involve any physical alteration to the plant. No new failure mechanisms will be introduced and the change does not affect the ability of the batteries to fulfill their safety-related function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(2) The separation of the DG start surveillance criteria into those criteria required to be met within 10 seconds, and those criteria required to be met following achievement of steady state conditions, does not involve any physical alteration to the plant. No new failure mechanisms will be introduced and the change does not affect the ability of the DGs to fulfill their safety-related function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The editorial changes are non-technical and therefore do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

(1) The allowance for unrestricted substitution of the modified performance discharge test in lieu of the service discharge test will not alter any accident analysis assumptions, initial conditions, or results. Consequently, it does not have any effect on the margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

(2) The proposed change to delete the requirement to demonstrate that the DG can achieve and maintain steady state operation within 10 seconds is not an accident analysis assumption. The accident analysis assumption that the DG reaches minimum conditions to accept load within 10 seconds is preserved. Consequently, it does not have any effect on the margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

(3) The editorial changes are non-technical and therefore 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.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 3, 1999.

Description of amendment request: The proposed changes will delete and/or relocate the additional primary-to-secondary leak rate limits and enhanced leakage monitoring requirements imposed following the 1987 steam generator tube rupture 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:

[O]peration of the North Anna Power Station in accordance with the proposed Technical Specification changes will not:

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

Eliminating the conservative primary-to-secondary leakage limits associated with the replaced steam generators and the operability requirements for the leakage monitoring instrumentation does not change the operation of the plant. The steam generators will be operated, inspected, and maintained in the same manner. No new accident initiators are established as a result of the proposed changes. Therefore, the probability of occurrence is not increased for any accident previously evaluated.

Removing the conservative primary-to-secondary leakage limits associated with the replaced steam generators and the operability requirements for the leakage monitoring instrumentation does not change the operation of the plant. Although the conservative leakage limits are being deleted, the remaining leakage limits will maintain the dose rate, in the event of a tube rupture, within the analyzed limits. Therefore, there is no increase in the consequences of any accident previously analyzed[.]

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

The proposed changes do not affect the operation of the plant. The steam generators will be operated, inspected, and maintained in the same manner. There are no modifications to the plant or steam generators as a result of the change. No new accident or event initiators are created by the removal of the conservative primary-to-secondary leakage limits associated with the replaced steam generators and the operability requirements for the leakage monitoring instrumentation. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type.

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

The proposed changes have no effect on any safety analyses assumptions. The remaining limits maintain primary-to-secondary leakage within the accident analysis assumptions. The proposed changes only eliminate overly conservative primary-to-secondary leakage requirements and the operability and surveillance requirements for the leakage monitoring system associated with the replaced steam generators. Therefore, the proposed changes do not result in 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia.

NRC Section Chief: Richard L. Emch, Jr.

Yankee Atomic Electric Co., Docket No. 50–029, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts

Date of amendment request: March 24, 1999.

Description of amendment request: The licensee submitted a request to delete License Condition 2.C.(10), which states: ''The licensee shall maintain a Fitness for Duty Program in accordance with the requirements of 10 CFR Part 26.''

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

The proposed change is administrative in nature in that it removes a reference in the YNPS Part 50 License to a regulatory requirement no longer applicable to a plant which has permanently ceased power operations and permanently removed fuel from its reactor vessel. This will permit more cost beneficial use of available resources with no diminution in the YNPS staff's ability to maintain the safe operation of the YNPS SFP [spent fuel pool]. The change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. Each potential accident in the YNPS FSAR [final safety analysis report] projects a maximum release of activity and no prompt mitigation actions. None of the analyzed scenarios resulted in a situation which could significantly [a]ffect the public health and safety. Removal of a regulatory requirement which does not apply to a plant which has permanently ceased power operations and permanently removed fuel from its reactor vessel cannot be deemed to involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different accident from any previously evaluated. The proposed change will not modify any plant systems or components and, therefore will not create the possibility of a new or different accident from any previously evaluated.

3. Involve a significant reduction in the margin of safety. Removal of a regulatory requirement which does not apply to a plant which has permanently ceased power operations and permanently removed fuel from its reactor vessel cannot be deemed to 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.

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Section Chief: Michael T. Masnik.

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.

Texas Utilities Electric Company, et al., Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: February 11, 1999.

Description of amendment request: The proposed amendments would credit soluble boron in the spent fuel pool water, in the maintenance of a subcritical condition, and allow an increase in spent fuel storage from 1291 to 2026 fuel assemblies.

*Date of publication of individual notice in **Federal Register***: May 12, 1999 (64 FR 25522).

Expiration date of individual notice: June 11, 1999.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P. O. Box 19497, Arlington, Texas.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: January 17, 1998, as supplemented by letters dated February 10, 1998, November 9, 1998, February 8, 1999, and February 26, 1999.

Brief description of amendment: This amendment authorizes changes to the Beaver Valley Power Station, Unit No. 1 (BVPS–1) Updated Final Safety Analysis Report (UFSAR). Specifically, the authorized changes to the UFSAR reflect revisions to the control room radiological dose calculations for the waste gas system line break accident analysis to correct a mathematical error discovered in a previous calculation, and use of more conservative assumptions in the revised analysis.

Date of issuance: May 12, 1999.

Effective date: As of the date of issuance.

Amendment No: 222.

Facility Operating License No. DPR–66. Amendment approved changes to the UFSAR.

Date of initial notice in ***Federal Register***: February 25, 1998 (63 FR 9601).

The February 10, 1998, November 9, 1998, February 8, 1999, and February 26, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

Date of application for amendment: August 23, 1996, as supplemented on April 9, 1999.

Brief description of amendment: The amendment revises Section 5.0, ''Design Features,'' and Section 6.0, ''Administrative Controls,'' of the Technical Specifications, adopting, for the most part, the format and content of the NUREG–1432, Revision 1, ''Standard Technical Specifications [STS] for Combustion Engineering Plants'' for the changes requested. This amendment also relocates certain portions of the design features section to other licensee-controlled documents in accordance with the STS.

Date of issuance: May 19, 1999.

Effective date: As of its date of issuance and shall be implemented within 30 days from the date of issuance: May 19, 1999.

Amendment No: 205.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: October 9, 1996 (61 FR 52965).

The April 9, 1999, letter provided clarifying information that did not change the scope of the original application and initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

Entergy Operations, Inc., Docket No. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of amendment request: December 19, 1996, as supplemented by letters dated August 6, 1998, and December 3, 1998.

Brief description of amendments: The amendments change requirements for the control room ventilation system for both Units 1 and 2.

Date of issuance: May 19, 1999.

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

Amendment Nos.: 196 and 206.

Facility Operating License Nos. DPR–51 and NPF–6: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: January 29, 1997 (62 FR 4348).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

Date of application for amendment: January 12, 1999, which superseded application dated May 31, 1996.

Brief description of amendment: The amendment adds an additional required action to the Limiting Condition for Operation (LCO) 3.9.1, ''Refueling Equipment Interlocks,'' of the Grand Gulf Technical Specifications. The additional action will allow an alternative to the current action for one or more inoperable refueling equipment interlocks. The current action is to ''suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).'' The alternative action will be to (1) insert a control rod withdrawal block, and (2) verify all control rods are fully inserted in core cells containing one or more fuel assemblies. The amendment also revised the Bases for LCO 3.9.1 actions to describe the alternative action.

Date of issuance: May 7, 1999.

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

Amendment No.: 138.

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

Date of initial notice in ***Federal Register***: February 10, 1999 (64 FR 6695), which superseded original notice of June 16, 1996 (61 FR 31178).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 7, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

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

Date of application for amendment: October 28, 1998, as modified by letter dated March 19, 1999.

Brief description of amendment: This amendment revises administrative requirements relating to: TS 6.5.1.6, Station Review Board Responsibilities; TS 6.8.4.d, Radioactive Effluent Controls Program; TS 6.10, Records Retention; TS 6.11, Radiation Protection Program; TS 6.12, High Radiation Area; and TS 6.15, Offsite Dose Calculation Manual.

Date of issuance: May 19, 1999.

Effective date: May 19, 1999.

Amendment No.: 231.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: December 18, 1998 (63 FR 64126).

The supplemental information contained clarifying information and did not change the initial proposed no significant hazards consideration determination and did not expand the scope of the application as described in the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

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

Date of application for amendment: May 5, 1998, as supplemented August 3 (2 letters), September 14, and December 22, 1998.

Brief description of amendment: The amendment approves the use of a small amount of containment overpressure to ensure sufficient net positive suction head for the emergency core cooling system pumps.

Date of Issuance: May 13, 1999.

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

Amendment No.: 206.

Facility Operating License No. DPR–16. Amendment authorizes changes to the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56250).

The supplemental letters provided additional information that was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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

Date of application for amendment: November 10, 1998.

Brief description of amendment: The proposed Technical Specification (TS) change would remove the restriction on the sale or lease of property within the exclusion area and replace the restriction with a requirement to retain complete authority to determine and maintain sufficient control of all activities including the authority to exclude or remove personnel and property within the minimum exclusion distance. A TS Bases page for the proposed change is included. Also included are clarifications and administrative changes which: (1) clarify TS definition 1.38 to become ''Site Boundary'' rather than the current term ''Exclusion Area'' to be consistent with the 10 CFR 20.1003 definition for Site Boundary and the 10 CFR 100.3 definition of Exclusion Area, (2) revise the TS definition from Exclusion Area to Site Boundary in TS 6.8.4(a)(9), and (3) revise and update the TS Table of Contents for Section I Definitions.

Date of Issuance: May 12, 1999.

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

Amendment No.: 205.

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

Date of initial notice in **Federal Register: December 12, 1998 (63 FR 66595).**

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 12, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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

Date of application for amendment: February 12, 1999.

Brief description of amendment: The amendment deletes the organizational chart and related references from the Appendix B Environmental Technical Specifications (ETS). In addition, the appearance and format of the ETS have been extensively revised.

Date of Issuance: May 18, 1999.

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

Amendment No.: 207.

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

Date of initial notice in **Federal Register: April 7, 1999 (64 FR 17026).**

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 18, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: January 18, 1999, as supplemented February 3 and March 17, 1999.

Brief description of amendment: The amendment removes Technical Specification (TS) 3/4.6.4.3, ''Containment Systems, Hydrogen Purge System,'' from the TS and allows downgrading the system to a non-safety-related system.

Date of issuance: April 12, 1999.

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

Amendment No.: 233.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 10, 1999 (64 FR 6704).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 12, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: January 18, 1999.

Brief description of amendment: The amendment modifies Technical Specification 3/4.2.2 to be in accordance with NRC-approved Westinghouse methodologies for the heat flux hot channel factor—$F_Q(Z)$. In addition, the amendment makes changes to the core operating limits and the analytical methods used to determine core operating limits contained in Section 6.9.1.6.a and b, respectively, by adding, modifying, or deleting references.

Date of issuance: May 10, 1999.

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

Amendment No.: 170.

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

Date of initial notice in **Federal Register**: February 10, 1999 (64 FR 6705).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 10, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, ThreeRivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut

PECO Energy Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania.

Date of application for amendments: October 30, 1998, as supplemented February 22, 1999.

Brief description of amendments: These amendments revised the overvoltage, undervoltage, and underfrequency allowable values associated with the reactor protection system monitoring channels and add supporting details to the Technical Specifications Bases 3/4.8.4.

Date of issuance: May 13, 1999.

Effective date: Units 1 and 2, as of date of issuance and shall be implemented within 30 days.

Amendment Nos.: 134 and 96.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (64 FR 64120)

The February 22, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

PECO Energy Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: January 12, 1999, as supplemented January 29 and March 10, 1999.

Brief description of amendment: This amendment revised Technical Specifications (TSs) Section 3/4.4.2, ''Safety/Relief Valves,'' and TS Bases Sections B 3/4.4.2, B 3/4.5.1 and B 3/4.5.2 to increase the allowable as-found main steam safety relief valve (SRV) code safety function lift setpoint tolerance from plus or minus 1% to plus or minus 3%. Also, the required number of operable SRVs in operational conditions 1, 2 and 3 will be increased from 11 to 12.

Date of issuance: May 17, 1999.

Effective date: May 17, 1999.

Amendment No.: 98.

Facility Operating License No. NPF–85. The amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9194)

The January 29 and March 10, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

PECO Energy Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: March 11, 1999, as supplemented April 21, 1999.

Brief description of amendment: The amendment revised the minimum critical power ratio safety limits and revised the associated Technical Specification Bases.

Date of issuance: May 14, 1999.

Effective date: As of the date of issuance and shall be implemented prior to restart following completion of the April 1999 refueling outage.

Amendment No.: 97.

Facility Operating License No. NPF–85. The amendment revised the Technical Specifications and/or License.

Date of initial notice in **Federal Register**: April 7, 1999 (64 FR 17028). The April 21, 1999, letter provided clarifying information that did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 14, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

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

Date of application for amendments: May 22, 1997, as supplemented by letters dated June 12, 1997, August 28, 1997, January 29, 1998, July 9, 1998, and March 12, 1999.

Brief description of amendments: The amendments authorize revisions to the licensing basis as described in the Final Safety Analysis Report (FSAR) Update to incorporate a modification to the Diablo Canyon Power Plant, Unit Nos. 1 and 2 component cooling water system.

Date of issuance: May 13, 1999.

Effective date: May 13, 1999, and shall be implemented in the next periodic update to the FSAR Update in accordance with 10 CFR 50.71(e).

Amendment Nos.: UNIT 1–134; UNIT 2–132.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Final Safety Analysis Report Update.

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40558).

The supplemental letters dated July 9, 1998, and March 12, 1999 provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

Date of application for amendments: May 8, 1996, as supplemented January 13, 1999.

Brief description of amendments: The amendments modified the technical specifications to allow refueling operation with 20 feet of water level in the refueling cavity for many operating conditions and at 12 feet of water level for certain specified conditions. The amendments also restored a phrase to a note to Limiting Conditions for Operation for TSs 3.9.4 and 3.9.5 that was inadvertently deleted by previous amendments.

Date of issuance: May 13, 1999.

Effective date: May 13, 1999, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2–153; Unit 3–144.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14285).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713.

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

Date of amendments request: April 2, 1999.

Brief Description of amendments: The amendment changes TS 3/4.4.9, ''Specific Activity,'' and the associated bases to increase the limit associated with dose equivalent iodine-131. The steady-state dose equivalent iodine-131 limit would be increased from 0.15μCurie/gram to 0.3 μCurie/gram and the transient limit for 80 percent to 100 percent power provided by Technical Specificaton Figure 3.4–1 will increase 9 μCurie/gram to 18 μCurie/gram with a corresponding increase in the 0 percent to 80 percent power limits.

Date of issuance: May 10, 1999.

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

Amendment Nos.: Unit 1–142; Unit 2–134.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

*Date of initial notice in **Federal Register***: April 8, 1999 (64 FR 17201).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 10, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama.

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

Date of amendment request: September 29, 1998.

Brief description of amendments: The amendments authorize the revision of the South Texas Project updated final safety analysis report (UFSAR) to incorporate the revised methodology to calculate the mass and energy release following a postulated large-break loss-of-coolant accident.

Date of issuance: May 20, 1999.

Effective date: May 20, 1999 Revisions will be incorporated into the next UFSAR update in accordance with the schedule in 10 CFR 50.71(e).

Amendment Nos.: Unit 1–110; Unit 2–97.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments authorize the revision of the UFSAR to incorporate the revised methodology.

*Date of initial notice in **Federal Register***: November 18, 1998 (63 FR 64123).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 20, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location:

Wharton County Junior College, J. M. Hodges.

Learning Center, 911 Boling Highway, Wharton, Texas 77488.

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: October 29, 1998, supplemented by letter dated March 15, 1999.

Brief description of amendments: The amendments relocate the requirements in Technical Specifications 3/4.7.9 and 6.10.3.l for snubbers to the Technical Requirements Manual.

Date of issuance: May 17, 1999.

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

Amendment Nos.: Unit 1–109; Unit 2–96.

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

*Date of initial notice in **Federal Register***: December 16, 1998 (63 FR 69346); renoticed April 7, 1999 (64 FR 17031).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 17, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

Wisconsin Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: February 15, 1999.

Brief description of amendment: This amendment revised Technical Specification Section 6, ''Administrative Controls,'' to reflect organizational changes, to relocate certain review and audit functions to the Operational Quality Assurance Program Description, and to eliminate redundant requirements.

Date of issuance: May 11, 1999.

Effective date: May 11, 1999.

Amendment No.: 145.

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

*Date of initial notice in **Federal Register***: April 7, 1999 (64 FR 17031).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 11, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001.

Dated at Rockville, Maryland, this 25th day of May 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–13765 Filed 6–1–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Consolidated Guidance About Materials Licenses: Program-Specific Guidance About 10 CFR Part 36 Irradiator Licenses, Dated January 1999

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is announcing the availability of NUREG–1556, Volume 6, ''Consolidated Guidance about Materials Licenses: Program-Specific Guidance about Part 36 Irradiator Licenses,'' dated January 1999.

ADDRESSES: Copies of NUREG–1556, Vol. 6, may be obtained by writing to the Superintendent of Documents, U.S. Government Printing Office, PO Box 37082, Washington, DC 20402–9328. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161. A copy of the document is also available for inspection and/or copying for a fee in the NRC Public Document Room, 2120 L Street, NW, (Lower Level), Washington, DC 20555–0001.

FOR FURTHER INFORMATION CONTACT: Ms. Sally L. Merchant, Mail Stop TWFN 9–F–31, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone: (301) 415–7874, E-mail: slm2@nrc.gov.

SUPPLEMENTARY INFORMATION: On April 23, 1998 (63 FR 20224), NRC announced the availability of draft NUREG–1556, Volume 6, ''Consolidated Guidance about Materials Licenses: Program-Specific Guidance about Self-Shielded Irradiator Licenses,'' dated March 1998, and requested comments on it. This draft NUREG report was the sixth program-specific guidance developed to support an improved materials licensing process. The NRC staff considered all the comments, including constructive suggestions to improve the document, in the preparation of the final NUREG report.