fellowships. The Board will meet in Boston, Massachusetts on June 15, 1995 from 9 a.m. to 4 p.m. The meeting of the Board is open to the public. The agenda includes discussion of the Institute's plans and priorities for program years 1995 and 1996; the status of new Board member nominations; and other relevant Institute matters. Records are kept of all Board proceedings and are available for public inspection at the National Institute for Literacy, 800 Connecticut Avenue, NW, Suite 200, Washington, DC 20006 from 8:30 a.m. to 5 p.m.

Dated: May 17, 1995.

Andrew J. Hartman,

Executive Director, National Institute for Literacy.

[FR Doc. 95–12536 Filed 5–22–95; 8:45 am]

BILLING CODE 6055–01–M

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 1, 1995, through May 12, 1995. The last biweekly notice was published on May 10, 1995 (60 FR 24904).

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 Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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 June 23, 1995, 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, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: April 6, 1995.

Brief description of amendments: The proposed amendment involves changes in personnel titles, implementation of line item improvements delineated in Generic Letter 93–07, ''Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans,'' changes in the Plant Review Board, and miscellaneous minor changes.

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

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

These changes involve (1) minor changes in the organization of PVNGS, (2) line item improvements recommended by the NRC, or (3) clarification or corrections to existing specifications. It is expected that the organizational changes will have a positive effect on the conduct of plant operations and safety-related work. Functions which are necessary to operate the facility safely and in accordance with the operating licenses, remain in the new organization. The line item improvements to the Technical Specifications will not affect the safe operation of the plant and continue to ensure proper control of administrative activities. The proposed changes will not affect the operation of structures, systems and components, and will not reduce programmatic controls such that plant safety would be affected. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not affect the operation of structures, systems and components, and will not reduce programmatic controls such that plant safety would be affected. The changes in the organization and as a result of line item improvements will continue to provide necessary oversight and control of administrative processes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

These changes are administrative and will not diminish any organizational or administrative controls currently in place. The proposed changes will not affect the operation of structures, systems and components, and will not reduce programmatic controls such that plant safety would be affected. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Project Director: William H. Bateman.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3, Maricopa County, Arizona

Date of amendment requests: April 18, 1995.

Description of amendment requests: The proposed Technical Specification amendments would revise the surveillance requirements for Technical Specification 3/4.4.4, ''Steam Generators,'' and the associated Bases. These amendments would allow the installation of tube sleeves as an alternative to plugging defective steam generator tubes.

Basis for proposed no significant hazards consideration determination:

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

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

The proposed amendment to permit the use of steam generator tube sleeves as an alternative to tube plugging is a safe and effective repair procedure that does not require removing a tube from service. Mechanical strength, corrosion resistance, installation methods, and inservice inspection techniques of sleeves have been shown to meet NRC acceptance criteria.

Analytical verifications were performed using design and operating transient parameters selected to envelope loads imposed during normal operating and accident conditions. Fatigue and stress analysis of sleeved tube assemblies were completed in accordance with the requirements of Section III of the ASME Code. The results of qualification testing, analysis and plant operating experience at other facilities demonstrates that the sleeving process is an acceptable means of maintaining steam generator tube integrity. The sleeve configuration has been designed and analyzed in accordance with the structural margins specified in Regulatory Guide (RG) 1.121. Furthermore, the installed sleeve will be monitored through periodic inspections on a sample basis with eddy current techniques. A sleeve-specific plugging margin, per the recommendations of RG 1.121, has been specified with appropriate allowances for NDE (nondestructive examination) uncertainty and defect growth rate.

The consequences of accidents previously analyzed are not increased as a result of sleeving activities. The hypothetical failure of the sleeve would be bounded by the current steam generator tube rupture analysis contained in the PVNGS (Palo Verde Nuclear Generating Station) UFSAR (updated final safety analysis report). Due to the slight reduction in diameter caused by the sleeve wall thickness, it is expected that the primary release rates would be less than assumed for the steam generator tube rupture analysis, and therefore would result in lower total primary fluid mass release to the secondary system. Additionally, further conservatism is introduced if the break were postulated to occur at a location on the tube higher than the location where a sleeve is installed. The overall effect would be reduced steam generator tube rupture release rates. The minimal reduction in flow area associated with a tube sleeve has no significant affect on steam generator performance with respect to heat transfer or system flow resistance and pressure drop. The installation of sleeves rather than plugging also maintains a greater heat transfer surface in the steam generator. In any case, the impacts are bounded by evaluations which demonstrate the acceptability of tube plugging which totally removes the tube from service. Therefore, in comparison to plugging, tube sleeving is considered a significant improvement with respect to steam generator performance. The cumulative impact of multiple sleeved tubes was evaluated to ensure the effects remain within the analytical design bases.

Recent industry experience with forced shutdown events associated with tube failures at sleeve junctions was assessed by ASP and ABB–CE. The root cause of these events has been attributed to the lack of proper post-installation stress relief and/or the imposition of high stresses due the tube growth restrictions at locked tube support. The material and design of the PVNGS steam generator supports minimizes the potential for locked supports. The tube supports are of eggcrate design and are constructed of ferritic stainless steel. The large flow area in the eggcrate design provides better irrigation and reduces the potential for steam blanketing, therefore, the tube-to-tube support crevices are less likely to be blocked by crud, boiler water deposits and corrosion products. Since the support material is type 409 ferritic stainless steel, it is not susceptible to magnetite corrosion which has resulted in denting and lockup at plants with carbon steel supports. These conclusions have been substantiated via tube pull activities conducted in PVNGS Unit 2. Although ABB–CE does not require post-weld heat treatment in all applications, APS will require that a post-weld stress relief be conducted for all sleeve installations.

APS has incorporated an integrated leakage monitoring program, utilizing equipment, procedure upgrades and administrative shutdown limits significantly lower than Technical Specification requirements. The program is designed to provide plant operators with the ability to detect and respond to changes in primary-to-secondary leakage and shutdown the unit prior to a significant leak or steam generator tube rupture, should sleeve or tube degradation exceed expected values. The program is designed to reduce the probability of steam generator tube rupture events.

Therefore, based on the above, the proposed amendment does not significantly increase 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 analyzed.

A sleeved steam generator tube performs the same function in the same passive manner as an unsleeved steam generator tube. Tube sleeves are designed, qualified, and maintained under the stress and pressure limits of Section III of the ASME Code and Regulatory Guide 1.121.

The installation of the sleeve, including weld and welder qualification and nondestructive examination (NDE), meets or exceeds the requirements of ASME Section XI. Three types of NDE are conducted. Ultrasonic Testing (UT) is performed to verify adequacy of the tube to sleeve weld assuring proper fusion. Eddy current testing (ET) is performed following each installation to establish baseline data for each sleeve in order to monitor future degradation of the primary to secondary pressure boundary. Visual inspections may be performed to verify or ascertain the mechanical and structural condition of a weld. Critical conditions which are checked include weld width and completeness, and the absence of visibly noticeable indications such as cracks, pits, and burn through.

ABB-Combustion Engineering Inc., Report CEN–613–P, ''Arizona Public Service Co., Palo Verde Units 1, 2, and 3, Steam Generator Tube Repair Using Leak Tight Sleeves,'' Revision 01, January 1995, demonstrates that the repair of degraded steam generator tubes using tube sleeves will result in tube bundle integrity consistent with the original design basis. An extensive analysis and corrosion and mechanical test programs were undertaken to prove the adequacy of tube sleeve repair. The proposed amendments have no significant effect on the configuration of the plant, and the change does not effect the way in which the plant is operated. Based upon the results of the analytical and test programs described in the ABB Combustion Engineering Inc. report, the tube sleeve fulfills its intended function and meets or exceeds established design criteria. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Evaluation of the sleeved tubes indicates no detrimental effects on the sleeve-tube assembly resulting from reactor system flow, coolant chemistries, or thermal and pressure conditions. Structural analyses of the sleeve-tube assembly, using demonstrated margins of safety, have established sleeve-tube integrity under normal and accident conditions. Structural analyses have been performed for sleeves which span the tube at the top of the tubesheet and which span the flow distribution plate or eggcrate support. Mechanical testing has been performed to support the analyses. Corrosion testing of typical sleeve-tube assemblies has been completed and reveals no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

Based upon the testing and analyses performed, the installation of tube sleeves will not result in a significant reduction in a margin of safety.

Steam generator tube integrity is maintained under the same limits for sleeved tubes as for unsleeved tubes, i.e., Section III of the ASME Code and Regulatory Guide 1.121. The portions of the installed sleeve assembly which represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. The degradation limit at which a sleeve/tube boundary is considered inoperable has been analyzed in accordance with Regulatory Guide 1.121 and is specified. Eddy current detectability of flaws has been verified by ABB Combustion Engineering. The Technical Specifications continue to require monitoring and restriction of primary to secondary system leakage through the steam generators. A conservative integrated leakage program employed by APS provides reasonable assurance than an orderly unit shutdown will occur prior to a significant increase in leakage due to failure of a sleeved or unsleeved tube. The minimal reduction in reactor coolant system flow, due to sleeving, is considered to have an insignificant impact on steam generator operation during normal operation or accident conditions and is bounded by tube plugging evaluations. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Project Director: William H. Bateman.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: February 16, 1993, as supplemented by letter dated May 2, 1995.

Description of amendment request: As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom Technical Specifications (TS) used.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG–0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG–0123.

The TSUP for Dresden and Quad Cities is not a complete adaption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

The February 16, 1993, and May 2, 1995, applications proposed to upgrade only Section 3/4.10 (Refueling Operations) of the Dresden and Quad Cities 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:

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

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.10 are based on STS guidelines or later operating BWR plant's NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems that make up the Refueling Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. The additional surveillance requirements improve the reliability and availability of all affected systems and therefore, reduce the consequences of any accident previously evaluated as the probability of the systems outlined within Section 3/4.10 of the proposed Technical Specifications, performing its intended function is increased by the additional surveillances.

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

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.10 is based on STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes, considering the acceptable operational modes in present specifications, the STS, or later operating BWRs. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications or adopt requirements that have been used successfully at other operating BWRs with designs similar to Dresden and Quad Cities. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that make up the Refueling Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the Refueling Systems are generally more prescriptive than the current requirements specified within the Technical Specifications; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the later individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain with their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 3/4.10 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden or Quad Cities Stations. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden and Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and Quad Cities and maintain necessary levels of system, component or parameter (reliability), the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the Refueling Systems when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

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

Date of amendment request: September 19, 1994, as supplemented by letter dated April 26, 1995.

Description of amendment request: The amendments would change the Technical Specifications (TS) to increase the enrichment limits for fuel stored in the fuel pools and establish restricted loading patterns and associated burnup criteria for qualifying fuel in the spent fuel pools. In addition, several administrative changes have been included in order to provide clarity to the TS and bring them more in line with the Standard Technical Specifications format. These changes are as follows:

(1) The TS index is changed to add TS 3/4.9.12 and 3/4.9.13, Tables 3.9–1 and 3.9–2 and Figure 3.9–1.

(2) TS 3/4.9.12, Spent Fuel Pool (SFP) Boron Concentration, is added to establish a boron concentration limit and to establish a Limiting Condition for Operation (LCO) for all modes of operation and to allow the numerical value of the limit to be specified in the Core Operating Limits Report (COLR).

(3) TS 3/4.9.13, Tables 3.9–1 and 3.9–2 and Figure 3.9–1 are being added to establish restricted loading patterns for spent fuel storage and associated burnup criteria.

(4) Corresponding BASES for TSs 3/4.9.12 and 3/4.9.13 are added to explain the basis for each LCO, Action Statement, and Surveillance Requirement covered by the subject TSs.

(5) TS 5.6, Fuel Storage, is changed to reflect limits for criticality analysis for fuel storage.

(6) TS 6.9, Reporting Requirements, is changed to reflect the inclusion of the SFP boron concentration limit values in the COLR as established by TS 3/4.9.12.

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

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

There is no increase in the probability or consequences of an accident in the new fuel vault since the only credible accidents for this area are criticality accidents and it has been shown that calculated, worst case K_{eff} for this area is (less than or equal to) 0.95 under all conditions.

There is no increase in the probability of a fuel drop accident in the Spent Fuel Storage Pool since the mass of an assembly will not be affected by the increase in fuel enrichment. The likelihood of other accidents, previously evaluated and described in Section 9.1.2 of the FSAR (Final Safety Analysis Report), is also not affected by the proposed changes. In fact, it could be postulated that since the increase in fuel enrichment will allow for extended fuel cycles, there will be a decrease in fuel movement and the probability of an accident may likewise be decreased. There is also no increase in the consequences of a fuel drop accident in the Spent Fuel Pool since the fission product inventory of individual fuel assemblies will not change significantly as a result of increased initial enrichment. In addition, no change to safety related systems is being made.

Therefore, the consequences of a fuel rupture accident remain unchanged. In addition, it has been shown that K_{eff} is (less than or equal to) 0.95, under all conditions. Therefore, the consequences of a criticality accident in the Spent Fuel Pool remain unchanged as well.

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

The proposed changes do not create the possibility of a new or different kind of accident since fuel handling accidents (fuel drop and misplacement) are not new or different kinds of accidents. Fuel handling accidents are already discussed in the FSAR for fuel with enrichments up to 4.0 weight %. As described in Section IV.9 of Attachment IV, additional analyses have been performed for fuel with enrichment up to 5.00 weight %. Worst case misloading accidents associated with the new loading patterns were evaluated. It was shown that the negative reactivity provided by soluble boron maintains K_{eff} (less than or equal to) 0.95.

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

The proposed change does not involve a significant reduction in the margin of safety since, in all cases, a K_{eff} [less than or equal to] 0.95 is being maintained. Criticality analyses have been performed which show that the new fuel storage vault will remain subcritical under a variety of moderation conditions, from fully flooded to optimum moderation. As discussed above, the Spent Fuel Pool will remain sufficiently subcritical during any fuel misplacement accident.

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: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

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: March 30, 1995, and supplemented May 5, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications (TS) by separation of the 24-hour emergency diesel generator (EDG) run from the hot restart EDG test.

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 TS changes would revise the EDG surveillance criteria to allow the EDG hot-start test with full ESF load acceptance to be performed separately and independently from the 24-hour EDG run. The proposed SRs (surveillance requirements) would continue to demonstrate that the objectives of these two tests are met. Specifically, the EDGs are shown to be: (1) Capable of starting and running continuously at full load capability for an interval not less than 24 hours, and (2) capable of restarting from a full load temperature condition. The proposed changes would not affect the EDGs' ability to support mitigation of the consequences of any previously evaluated accident. Additionally, the proposed changes to the SRs do not affect the initiating assumptions or progression of any accident sequence.

Therefore, operation of the facility would not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed TS SR changes do not require any physical changes to the plant or equipment, and do not impact any design or functional requirements of the EDGs. The proposed changes do not create any plant configurations which are prohibited by the TS. The proposed changes continue to meet the EDG test objectives associated with demonstrating EDG operability.

Therefore, 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.

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

The proposed TS SR changes do not require any physical changes to the plant or equipment and do not impact any design or functional requirements of the EDGs. Surveillance testing in accordance with the proposed TS will continue to demonstrate the ability of the EDGs to perform their intended function of providing electrical power to mitigate design basis transients, consistent with the plant safety analyses.

Therefore, operation of the facility in accordance with the proposed amendments would not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of § 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: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Project Director: David B. Matthews.

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

Date of amendment request: April 7, 1995.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to relocate the axial power distribution limits to the Core Operating Limits Report (COLR).

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

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

The proposed change relocates the cycle-specific Axial Power Distribution (APD) limits contained in Figure 1–2 of the Technical Specifications (TS), to the Core Operating Limits Report (COLR). This change is consistent with the NRC recommendations of Generic Letter 88–16, and will not modify the methodology used in generating the limits nor the manner in which they are implemented. The methodology used to determine the APD limits is reviewed and approved by the NRC in accordance with TS 5.9.5. The APD limits will continue to be determined by analyzing the same postulated events as previously analyzed. The plant will continue to operate within the limits specified in the COLR and will take the same remedial actions if the APD limit is exceeded as required by the current TS. Therefore, the proposed change would not increase 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.

There will be no physical alterations to the plant configuration, changes to setpoint values, or changes to the implementation of setpoints or limits as a result of this proposed change. The proposed change only relocates the APD figure from the TS to the COLR consistent with NRC Generic Letter 88–16. Therefore, the proposed change does 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.

As indicated above, the implementation of the APD into the COLR, consistent with the guidance of NRC Generic Letter 88–16, makes use of the existing safety analysis methodologies and the resulting limits and setpoints for plant operation. Additionally, the safety analysis acceptance criteria for operations with the proposed change have not changed from that use in the current reload analysis. Therefore, the margin of safety is not reduced due to the relocation of the APD from the TS and implementation in the COLR.

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: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

Attorney for licensee: LeBoeuf, Lamb, Leiby, and MacRae, 1875 Connecticut Avenue, NW., Washington, DC 20009–5728.

NRC Project Director: William Bateman.

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 amendment requests: April 19, 1995 (Reference LAR 95–03).

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 to revise TS 3/4.8.1.1, ''A.C. Sources, Operating.'' The specific TS changes proposed are as follows:

(1) TS 4.8.1.1.2b.8), emergency diesel generator (EDG) 24-hour load run and hot restart surveillance, would be revised to delete the requirement to perform TS 4.8.1.1.2b.5)b), loss of offsite power (LOOP) load sequencing surveillance within 5 minutes following the 24-hour test.

(2) New TS 4.8.1.1.2e. would be added to perform an EDG hot restart test within 5 minutes of shutting down the EDG after the EDG has operated for at least 2 hours at a load of greater than or equal to 2484 kW.

(3) TS 4.8.1.1.2b.8), TS 4.8.1.1.2e., and footnote ''*'' on page 3/4 8–5 would be changed to be cycle-specific with the new TS requirements effective for Units 1 and 2, Cycle 8 and after.

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

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

Demonstrating emergency diesel generator (EDG) hot restart capability without sequencing loss of offsite power (LOOP) loads does not invalidate or reduce the effectiveness of the hot restart test, since normal operating temperatures are achieved prior to the hot restart test. Sequencing the LOOP loads does not contribute to verifying that the EDG will start from normal operating temperatures. The proposed TS 4.8.1.1.[2]e may be performed in any plant condition since performance of this new surveillance will have no adverse effect on plant operations. The reliability of the EDGs is not affected by the proposed changes.

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

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

The proposed changes do not involve any physical alterations to the plant. The proposed changes will not have any adverse effect on the ability of the EDGs to perform their required safety function.

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

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

The proposed changes will not alter any accident analysis assumptions, initial conditions, or results. Consequently, the proposed changes do not have any effect on the margin of safety. The proposed changes to the surveillance requirements would continue to demonstrate the ability of the EDGs to perform their intended safety function.

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

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

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

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, PO Box 7442, San Francisco, California 94120.

NRC Project Director: William H. Bateman.

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: November 21, 1994.

Description of amendment request: The proposed change would extend the Type A test (i.e., Containment Integrated Leak Rate Test (CILRT)) interval on a one-time 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 Technical Specifications (TS) change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The accidents which are potentially adversely impacted by the proposed change are any Loss of Coolant Accident (LOCA) inside primary containment as described in the PBAPS, Units 2 and 3 UFSAR.

The proposed change increases the surveillance interval of the 10 CFR part 50, appendix J Type A test (i.e., Containment Integrated Leakage Rate Test (CILRT)) from 46 months to 70 months. This test is performed to determine that the total leakage from containment does not exceed the maximum allowable primary containment leakage rate (i.e., designated La) at a calculated peak containment internal pressure (Pa), as defined in 10 CFR part 50, appendix J. The primary containment limits the leakage of radioactive material during and following design bases accidents in order to comply with the offsite does limits specified in 10 CFR part 100. Accordingly, the primary containment is not an accident initiator. It is an accident mitigator. No physical or operational changes to the containment structure, plant systems, or components would be made as a result of the proposed change. Therefore, the probability of occurrence of an accident previously evaluated is not increased.

The failure effects that are potentially created by the proposed one-time TS change have been considered. The relevant components important to safety which are potentially affected are the containment structure, plant systems, and containment penetrations. There are no physical or operational changes to any plant equipment associated with the proposed TS change. Therefore, the probability or consequences of a malfunction of equipment important to safety is not increased.

The proposed change introduces the possibility that primary containment leakage in excess of the allowable value (i.e., La) would remain undetected during the proposed 24 month extension of the interval between the Type A tests. The types of mechanisms which would cause degradation of the primary containment can be categorized into two types. These are: (1) Degradation due to work which is performed as part of a modification or maintenance activity on a component or system (i.e., activity-based), or; (2) degradation resulting from a time-based failure mechanism.

A review of the history of the PBAPS, Unit 3 CILRT results was performed to evaluate the risk of activity-based and time-based degradation. This review has determined that the potential for a time-based and activity-based failure is minimal. The PBAPS LLRT program would identify most types of penetration leakage. The LLRT program involves measurement of leakage from Type B and Type C primary containment penetrations as defined in 10 CFR part 50, appendix J.

The 10 CFR part 50, appendix J, Type B tests are intended to detect local leaks and to measure leakage across pressure containing or leakage-limiting boundaries other than values, such as containment penetrations incorporating resilient seals, gaskets, expansion bellows, flexible seal assemblies, door operating mechanism penetrations that are part of the containment system, doors, and hatches. 10 CFR part 50, appendix J, Type C testing is intended to measure reactor system primary containment isolation valve leakage rates. The frequency of the Type B and Type C testing is not being altered by the proposed TS change. The acceptance criterion for Type B and Type C leakage is 0.6 La (i.e., 0.3% wt/day) which, when compared to the Type A test acceptance criterion of 0.75 La (i.e., 0.375% wt/day), is a significant portion of the Type A test allowable leakage.

The proposed TS change only extends the interval between two consecutive Type A tests. The Type B and Type C tests will be performed as required. The Type B and Type C tests will continue to be used to confirm that the containment isolation valves and penetrations have not degraded. Containment system components that would not be tested are the containment structure itself and small-diameter instrumentation lines. Time-based degradation of any of the instrumentation lines would not likely be identified by faulty instrument indication or during instrument calibrations that will be performed during the PBAPS, Unit 3 refueling outage 10. In examining the potential for a time-based failure mechanism that could cause significant degradation of the containment structure, we concluded that the risk, if any, of such a mechanism is small since the design requirements and fabrication specifications established for the containment structure are in themselves adequate to ensure containment leak tight integrity.

Based on the above evaluation, we have concluded that the proposed TS change will have a negligible impact on the consequences of any accident previously evaluated.

Although this review concluded that the risk of undetected primary containment degradation is not increased, the Individual Plan Examination (IPE) for PBAPS, Units 2 and 3, was also reviewed in order to access the impact of exceeding the primary containment allowable leakage rate, if a non-mechanistic activity type (i.e., time-based) failure were to occur. The IPE included an evaluation of the effect of various containment leakage sizes under different scenarios. The IPE results showed that a containment leakage rate of 35% wt/day would represent less than a 5% increase in risk to the public of being exposed to radiation. This evaluation was based on a study performed by Oak Ridge National Laboratory for light water reactors that evaluated the impact of leakage rates on public risk. As stated earlier, the current value of La for PBAPS, Unit 3, is 0.5% wt/day, which is significantly less than the 35% wt/day discussed in the IPE evaluation.

Therefore, the proposed TS change involving a one-time extension of the Type A test interval and performing the Type A test after the second appendix J 10-year service period will not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is an increase of a surveillance test interval and does not make any physical or operational changes to existing plant systems or components. Primary containment acts as an accident mitigator not initiator. Therefore, the possibility of a different type of accident than any previously evaluated or the possibility of a different type of equipment malfunction is not introduced.

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

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

The total primary containment leakage rate ensures that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure. As an added conservatism, the measured overall leakage rate is further limited to less than or equal to 0.75 La during performance of periodic tests to account for possible degradation of the containment leakage barriers between leakage tests. There is the potential that containment degradation could remain undetected during the proposed 24 month surveillance interval extension and result in the containment leakage exceeding this allowable value assumed in safety analysis. A review of the history of the PBAPS, Unit 3 CILRT results was performed to evaluate the risk of activity-based and time-based degradation. This review has determined that the potential for a time-based and activity-based failure is minimal. The PBAPS LLRT program would identify most types of penetration leakage. The LLRT program involves measurement of leakage from Type B and Type C primary containment penetrations as defined in 10 CFR part 50, appendix J.

The 10 CFR part 50, appendix J, Type B tests are intended to detect local leaks and to measure leakage across pressure containing or leakage-limiting boundaries other than valves, such as containment penetrations incorporating resilient seals, gaskets, expansion bellows, flexible seal assemblies, door operating mechanism penetrations that are part of the containment system, doors, and hatches. 10 CFR part 50, appendix J, Type C testing is intended to measure reactor system primary containment isolation valve leakage rates. The frequency of the Type B and Type C testing is not being altered by the proposed TS change.

Therefore, we have concluded that the proposed extended test interval would not result in a non-detectable PBAPS, Unit 3 primary containment leakage rate in excess of the allowable value (i.e., 0.5% wt/day) established by the TS and 10 CFR part 50, appendix J.

Therefore, the proposed TS 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: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

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

NRC Project Director: John F. Stolz.

Public Service Electric & Gas Company, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: April 4, 1995.

Description of amendment request: The amendment would provide a one-time interval extension for the Type A test required by 10 CFR part 50, appendix J. The extension would allow the test to be conducted during the thirteenth refueling outage, rather than the twelfth refueling outage.

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

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

The proposed change will provide a one-time exemption from 10 CFR part 50, appendix J Section III.D.1(a) leak rate test schedule requirement. This change will allow for a one-time test interval for Type A Integrated Leak Rate Tests (ILRTs) of 65+/−10 months.

Leak rate testing is not an initiating event in any accident, therefore, this proposed change does not involve a significant increase in the probability of a previously evaluated accident.

Type A tests are capable of detecting both local leak paths and gross containment failure paths. The history at Salem Generating Station Unit 1 (SGS1) demonstrates that Type B and C Local Leak Rate Tests (LLRTs) have consistently detected any excessive local leakages. SGS1 has passed all of its ILRTs with significant margin.

Administrtive controls govern the maintenance and testing of containment penetrations such that the probability of excessive penetration leakage due to improper maintenance or valve misalignment is very low. Following any maintenance that could affect the leakage characteristics of any containment penetration, an LLRT is performed to ensure acceptable leakage levels. Following any LLRT on a containment isolation valve, an independent valve alignment check is performed before declaring the penetration OPERABLE. Therefore, Type A testing is not necessary to ensure acceptable leakage rates through containment penetrations.

While Type A testing is not necessary to ensure acceptable leakage rates through containment penetrations, Type A testing is necessary to demonstrate that there are no gross containment failures. Structural failure of the containment is considered to be a very unlikely event, and in fact, since SGS1 has been in operation, it has never failed a Type A ILRT. Therefore, a one-time exemption increasing the interval for performing an ILRT does not result in a significant decrease in the confidence in the leak tightness of the containment structure.

Therefore, this proposed change does not result in a significant increase of the probability or consequences of any previously evaluated accident.

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

This proposed change allows a one-time interval of 65+/−10 months for the next ILRT. The method of performing the test is not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. A one-time extension of the ILRT test interval has no influence on, nor does it contribute in any way to, the possibility of a new or different kind of accident or malfunction from those previously analyzed.

3. Will not involve a significant reduction in a margin of safety.

The purpose of the existing schedule of ILRTs is to ensure that the release of radioactive materials will be restricted to those leak paths and leak rates assumed in accident analyses. The relaxed schedule for ILRTs does not allow for relaxation of Type B and C LLRTs. Therefore, methods for detecting local containment leak paths and leak rates are unaffected by this proposed change. Given that the test history for ILRTs shows no failure during plant life, a one-time increase of the test interval does not lead to a significant probability of creating a new leakage path or increased leakage rates, and the margin of safety inherent in existing accident analyses is maintained. Therefore, this change does not involve a significant reduction kin 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: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

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

NRC Project Director: John F. Stolz.

Public Service Electric & Gas Company, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: May 4, 1995.

Description of amendment request: The amendment would authorize a one-time extension for the Type A test (overall integrated containment leakage rate) that is required by 10 CFR part 50, appendix J. The current Technical Specification would require that this test be conducted by July 7, 1995. The amendment would allow this test to be conducted by November 30, 1995.

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

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

The proposed change involves no hardware changes, no changes to the operation of any systems or components, and no changes to existing structures. This change is temporary, allowing a one-time extension of a specific surveillance requirement for cycle 12 to allow surveillance testing to coincide with the twelfth refueling outage. The proposed surveillance interval extension is short and will not result in any significant reduction in structural reliability nor will the extension affect the ability of the structure in performing its intended functions. to preclude the possibility of an undetected containment failure/leakage at a valve or penetration seal, Type ''B'' and ''C'' tests will continue to be performed as required by the Technical Specifications. Therefore, this change will not involve a significant increase in the probability or consequences of any accidents previously evaluated.

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

Extending the surveillance interval for the performance of specific testing will not create the possibility of any new or different kinds of accident. No changes are required to any system configurations, plant equipment, or analyses. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will not involve a significant reduction in a margin of safety.

The proposed change will not alter any assumptions, initial conditions, or results of any accident analyses. The safety limits assumed in the accident analyses and the design function of the structure required to mitigate the consequences of any postulated accidents will not be changed since only the surveillance interval is being extended. Historical performance indicates a high degree of reliability, and surveillance testing performed during continued plant operation will verify that Salem 1 will remain within analyzed limits. Consequently, the 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, New Jersey 08079.

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

NRC Project Director: John F. Stolz.

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 18, 1995.

Description of amendment request: The amendments would delete the quarterly leak rate test for the containment pressure-vacuum relief valves which is presently required because of the valves' resilient seat material. The resilient valve seat material will be replaced with a hard seat (metal to metal) design. The valves would still remain in the 10 CFR part 50 appendix J, Type C leak rate test program.

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

The containment pressure/vacuum relief valves are normally closed, and are used under administrative control to maintain containment internal pressure within −1.5 psig and +0.3 psig, as required by SGS Technical Specifications. The pressure/vacuum relief valves are relied upon for containment isolation and automatically close on high containment pressure or high containment atmosphere radioactivity. The pressure/vacuum relief system does not affect the probability of any previously evaluated accident.

The containment isolation function of the pressure/vacuum relief valves limits the consequences of a radiological release inside containment (i.e., Loss of Coolant Accident). The proposed changes to eliminate quarterly pressure drop (leak rate) testing would not increase the consequences of any previously evaluated accident. The valve flow characteristics and closure time requirements are not affected. The valves will continue to be subject to the Type C leak rate test criteria of 10 CFR part 50, appendix J. The deletion of the augmented quarterly test requirement is justified by replacement of the resilient valve seat material (which has a history of degradation and loss of leaktightness) with a metal to metal seating design.

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

Eliminating quarterly leak rate testing based on improved valve design would not result in any new or different kind of accident. The valves would continue to perform the containment isolation function consistent with the plant safety analyses, and would not adversely affect the initiation or progression of any accident sequence.

(3) Do not involve a significant reduction in a margin of safety.

This proposal involves replacement of the existing pressure/vacuum relief valves, which have resilient seating material, with valves using a hard seat (metal to metal design). Based on the improved design and operating experience of the replacement valves, augmented quarterly leak rate testing is no longer necessary or appropriate to verify leaktightness of the valves. Periodic leak rate testing will continue to be performed in accordance with 10 CFR part 450, appendix J. The pressure/vacuum relief valves will continue to maintain their containment isolation capability such that no margin of safety is affected by the proposed changes.

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, New Jersey 08079.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: May 3, 1995 (TS 93–09).

Descripton of amendment request: The proposed change would revise the implementation schedule for Amendment Nos. 182 and 174 from that stated in the amendments when they were approved by the Commission by letter dated May 24, 1994. As issued, the amendments reflected the licensee's plans to implement the changes for both units during the Unit 2 Cycle 6 refueling outage. However, by letter dated August 19, 1994, the licensee requested that implementation be delayed to 1995. This request was granted by Amendment Nos. 188 and 180 for Units 1 and 2 respectively by letter dated October 17, 1994. By letter dated May 3, 1995, the licensee informed the staff that evaluation of the design changes have concluded that significant safety risks would be involved with modification activities associated with installation. Therefore, the licensee has requested that implementation of the amendment be changed to specify that the amendment will be implemented along with the related plant modifications, without specifying the date when the modifications would be performed. No changes to the technical specification pages other than those approved when the amendments were issued are needed.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has determined that the no significant hazards consideration exists. This analysis was provided in the original submittal for the amendment from the licensee dated October 1, 1993, and was used in the preparation of the amendments. The licensee has determined that this analysis remains valid for the proposed revision and that the changes do not constitute a significant hazard. The staff previously issued the proposed finding in the **Federal Register** (59 FR 4947 and 59 FR 47182) and there were no public comments on the finding. This analysis is reproduced as follows:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed revision supports the implementation of design logic and setpoint changes to the loss-of-power relaying. This relaying is designed to ensure adequate voltage is available to safety-related loads in order to enhance their operability and support accident mitigation functions and to provide for auxiliary feedwater (AFW) pump starts. The design changes alter relay logic and delete unnecessary relaying, but do not change the diesel generator (D/G) start and load-shedding actuations that result from loss-of-power conditions. Therefore, no new actuations or functions have been created; and because the existing and proposed functions provide for accident mitigation considerations that are not the source of an accident, the probability of an accident is not increased. The deletion of the 6.9-kilovolt shutdown board normal-feeder undervoltage relays actually reduces the potential for inadvertent shutdown board blackouts as a result of short-duration voltage transients or instrument failures.

The setpoints and time delays for loss-of-power functions have been modified based on the guidelines developed by the Electrical Distribution System Clearinghouse as evaluated and determined through detailed analysis by TVA. This design is documented in TVA Calculations SQN–EEB–MS–TI06–0008, 27DAT, and DS–1–2 and is available for NRC review at the SQN site. The assigned values are conservative settings that will ensure adequate voltage is supplied to safety-related loads for accident mitigation and safety functions under normal, degraded, and loss-of-offsite-power voltage conditions with appropriate time delays to prevent damage to electrical loads and minimize premature or unnecessary actuations. The identification of loss-of-voltage conditions is enhanced by the design changes to ensure the timely sequencing of loads onto the D/G and the initiation of AFW pump starts for accident mitigation. Because there are no reductions in safety functions resulting from the design logic, setpoint, and time-delay changes to the loss-of-power instrumentation and offsite dose levels for postulated accidents will not be increased, the consequences of an accident are not increased.

The applicable mode addition, TS 3.0.4 exclusion deletion, and response time measurement clarification incorporated in the proposed change do not affect plant functions. These changes reflect the requirements that SQN has been maintaining and serve to clarify the requirements to provide consistency of application and easier understanding. The AFW footnote addition and bases revision only clarify operability conditions that are consistent with the plant design for the AFW pump and loss-of-power instrumentation. Because there are no changes to plant functions or operations, these revisions have no impact on accident probabilities or consequences.

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

As described above, the loss-of-power instrumentation ensures adequate voltage to safety-related loads by initiating D/G starts and load shedding and provides for AFW pump starting, but is not considered to be the source of an accident. Although the design logic, setpoint, and time-delay actuation criteria have changed, the output functions to various plant systems that actuate for load shedding and D/G starts remain the same. Therefore, actuation criteria have been affected, but not safety functions, and the TVA evaluation has confirmed that the new design enhances the ability to maintain adequate voltage to support safety functions. Since safety functions have not changed and the new loss-of-power instrumentation design continues to support operability of safety-related equipment, no new or different accident is created.

The applicable mode addition, TS 3.0.4 exclusion deletion, and response time measurement clarification, as well as the AFW operability clarifications, do not affect plant functions and will not create a new accident.

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

The proposed loss-of-power TS changes support design logic, setpoint, and time-delay requirements that have been verified by

TVA analysis to provide acceptable voltage levels for safety-related components. In determining the acceptability of these voltage levels, the minimum voltage for operation as well as detrimental component heating resulting from sustained degraded-voltage conditions were considered. This design ensures that safety-related loads will be available and operable for normal and accident plant conditions. The applicable mode addition, TS 3.0.4 exclusion deletion, response time measurement clarification, and AFW operability clarifications provide enhancements to TS requirements and do not affect plant functions. Therefore, no safety functions are reduced by these changes and there is no reduction in the margin of safety.

The NRC 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: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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

NRC Project Director: Frederick J. Hebdon.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: April 28, 1995.

Description of amendment request: The proposed amendment would extend for one more operating cycle an exception to Limiting Condition for Operation (LCO) 3.0.4 as it applies to the Technical Specification for the main steam isolation valve leakage control system. The existing LCO 3.0.4 exception was issued by Amendment 63 to the Operating License, and will expire upon completion of the fifty cycle of plant operation. The extension is proposed for the duration of the sixth cycle of operation to permit completion of activities necessary to implement the most appropriate permanent resolution for the issue of secondary containment bypass leakage through the main steam line drains.

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.

This License Amendment application proposes an extension for one operating cycle of the exception to Limiting Condition for Operation for Operation (LCO) 3.0.4 as it applies to the Technical Specification for the MSIV [main steam isolation valve] Leakage Control system. This extension is proposed for the duration of the sixth cycle of PNPP (Perry Nuclear Power Plant) operation, to permit completion of activities necessary to implement the most appropriate permanent resolution for the issue of secondary containment bypass leakage through the Main Steam Line drains. During the sixth cycle, the drains will remain in their current configuration, which seals off the bypass leakage path. The sealed drain path results in a temporary inoperability of the Inboard MSIV Leakage control system (MSIV–LCS) subsystem when the plant is operated below 50% power, due to condensate build-up in the bottom of the steam lines between the MSIVs. The requested 3.0.4 exception is necessary to permit plant startups with this temporary inoperability, for the duration of the sixth operating cycle.

The probability of occurrence of a previously evaluated accident is not affected by the proposed extension of the LCO 3.0.4 exception since no change to the plant or to the manner in which the plant is operated is involved. The existing plant configuration will be maintained for another operating cycle, and possible concerns resulting from that configuration have been analyzed. The extra weight of the water pooled between the MSIVs was analyzed with respect to piping supports and seismic considerations and was found to be acceptable, and any condensate that is carried past the outboard MSIVs will be drained to the condenser by drain connections downstream of the outboard MSIVs before it can reach the turbine. The temporary inoperability of the Inboard MSIV–LCS when below 50% power has no impact on accident initiation probability, since LCS does not serve to prevent accidents, but is only used in mitigating the consequences of Loss of Coolant Accidents that have already occurred.

The consequences of an accident are not significantly increased in that the Outboard MSIV–LCS will be available to perform the MSIV–LCS function by mitigating the consequences of a Loss of Coolant Accident (LOCA) during the temporary period in which the Inboard MSIV–LCS is unavailable. Any condensate that is carried past the outboard MSIVs will be drained to the condenser by drain connections downstream of the outboard MSIVs; therefore no impairment of the Outboard MSIV–LCS will result from condensed water.

The Action statement for one inoperable LCS subsystem remains the same, and the limits plant operation to the previously established 30-day Allowable Outage Time. The Action required if both the subsystems of MSIV–LCS were to become inoperable also remains the same. The MSIV function of isolating the Main Stream Lines is also unaffected by the existing plant configuration, since MSIV performance will not be affected by the existence of accumulated water in the bottom of the steam lines between the MSIVs during the plant operation below 50% power. Therefore, if necessary, the Main Steam Lines will be isloated, and leakage past the MSIVs will be routed for filtration as in the design-basis radiological analyses, and the consequences of previously evaluated accidents will remain unaffected.

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 to permit inoperability of the Inboard MSIV–LCS during periods of startup and power ascension to 50% RTP (rated thermal power) and during shutdown below 50% RTP does not create the possibility of a new or different kind of accident from any previously evaluated. The Inboard MSIV–LCS is only credited during a Recirculation Line Break LOCA wherein Reactor Coolant System depressurization occurs. The temporary unavailability of the Inboard MSIV–LCS. the amendment to the Technical Specifications is an administrative change that does not involve any change to the current plant design or methods of operation. No new plant equipment failure modes or accident initiators are introduced by the extension of the LCO 3.0.4 exception.

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

The response to the Recirculation Line Break LOCA will not be significantly affected since the Outboard MSIV–LCS can be assumed to be available. Allowing entry into Operational Conditions 1, 2 and 3 while utilizing the existing Action statement does not significantly reduce the margin of safety since the duration of time allowed for remaining in that Action statement is not increased. The proposed change will have no adverse impact on the reactor coolant system pressure boundary nor will any other system protective boundary or safety limit be affected.

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: Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Gail H. Marcus.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: May 1, 1995.

Description of amendment request: The proposed amendment would eliminate selected response time testing requirements, and incorporate guidance provided by Generic Letter 93–08, ''Relocation of Technical Specification Tables of Instrument Response Time Limits.''

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

For those proposed changes dealing with the elimination of selected response time test requirements, the purpose of the proposed Technical Specification change is to eliminate response time testing requirements for selected components in the Reactor Protection System, Isolation system, and Emergency Core Cooling System. The BWR Owners' Group has completed an evaluation which demonstrates that the response time testing is redundant to other Technical Specification required testing. These other tests, in conjunction with actions taken in response to NRC Bulletin 90–01, ''Loss of Fill-Oil in Transmitters Manufactured by Rosemount,'' and Supplement 1, are sufficient to identify failure modes or degradations in instrument response time and ensure operation of the associated systems within acceptable limits. There are no known failure modes that can be detected by response time testing that cannot also be detected by the other required Technical Specification testing. This evaluation was documented in NEDO–32291, ''System Analyses for Elimination of Selected Response Time Testing Requirements,'' January 1994, and the letter from T. Green to P. Loeser dated April 15, 1994 which were approved by an NRC Safety Evaluation dated December 28, 1994. The applicability of this evaluation to the Perry Nuclear Power Plant (PNPP) has been confirmed. In addition, PNPP will complete the additional actions identified in the NRC staff's Safety Evaluation of NEDO–32291.

Because of the continued application of other existing Technical Specification required tests such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, the response times of these systems will be maintained within the acceptance limits assumed in plant safety analysis and required for successful mitigation of an initiating event. The proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended function within their required response time, nor do the proposed changes themselves affect the operation of any equipment. As a result the proposed changes dealing with elimination of selected response time tests do not involve a significant increase in the probability or the consequences of an accident previously evaluated.

For those changes dealing with moving the surveillance requirement for ECCS RESPONSE TIME testing from the instrumentation section to the system section of the Technical Specifications, no change in testing requirements (other than the elimination of the instrument loops implemented as part of the NEDO–32291 changes) has been introduced. The relaxation in Applicability does not increase the probability or the consequences of an accident previously evaluated, since there are no design basis events during OPERATIONAL CONDITION 4 and 5 where ECCS systems are relied upon.

For those changes dealing with relocation of the response time limits from Technical Specification Tables and into the Updated Safety Analysis Report (USAR), the proposed changes are administrative in nature in that the test requirements and time limits are still requirements, but the placement of the limits have been relocated from the Technical Specifications and into the USAR. Therefore these changes do not involve a significant increase in the probability or the consequences of an accident previously evaluated.

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

None of the proposed Technical Specification changes affect the capability of the associated systems to perform their intended function within the acceptance limits assumed in plant safety analyses and required for successful mitigation of an initiating event. The proposed changes also do not change the manner in which any plant equipment is operated. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The current Technical Specification response times are based on the maximum allowable value assumed in the plant safety analyses. These analyses conservatively establish the margin of safety. As described above, the proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended function within the allowed response time used as the basis for the plant safety analyses. Plant and system response to an initiating event will remain in compliance within the assumptions of the safety analyses, and therefore the margin of safety is not affected.

Although not explicitly evaluated, the proposed Technical Specification changes dealing with response time testing elimination will provide an improvement to plant safety and operation by reducing the time safety systems are unavailable, reducing safety system actuation, reducing plant shutdown risk, limiting radiation exposure to plant personnel, and eliminating the diversion of key personnel to conduct unnecessary testing. Therefore, the proposed changes do not result in a significant reduction in a margin of safety, and may result in an overall increase in the margin of safety.

The changes dealing with relocation of the time response limits from the Technical Specifications to the USAR is an administrative change that does not affect either the requirements to perform response time testing or the limits associated with the response time tests. Future changes to the limits will be controlled by 10 CFR 50.59. Therefore, this portion of the change does not result in a significant decrease 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: Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: April 26, 1995.

Description of amendment request: the proposed amendment would revise Technical Specification (TS) Surveillance Requirements 3/4.7.6 and associated Bases to reduce the upper limit on the control room filtration subsystem flow rate. It would also adopt ASTM D–3803–1989 as the laboratory testing standard for control room filtration and control building pressurization charcoal absorber.

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 revision does not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

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

Overall protection system performance will remain within the bounds of the accident analysis documented in FSAR Chapter 15 * * * since no hardware changes are proposed.

The Control Room Emergency Ventilation System (CREVS) will continue to function in a manner consistent with the above analysis assumptions and the plant design basis. There will be no degradation in the performance of or an increase in the number of challenges to equipment assumed to function during an accident situation.

These Technical Specification revisions do not involve any hardware changes nor do they affect the probability of any event initiators. The change to the control room filtration flow rate is consistent with the original licensing basis and will ensure an average atmosphere residence time of greater than or equal to 0.25 sec. There will be no change to ESF (engineered safety feature) actuation setpoints or accident mitigation capabilities. The laboratory testing will demonstrate the required absorber performance after a design basis LOCA (loss-of-coolant accident).

The control room dose analyses assume a total flow rate through the control room filtration units that is less than the proposed upper limit. As such, there will be no changes required to the control room dose analyses.

Based on the above, these Technical Specification changes will not increase the probability or consequences of an accident or malfunction.

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

As discussed above, there are no hardware changes associated with these Technical Specification revisions nor are there any changes in the method by which any safety-related plant system performs its safety function.

Revisions to the Surveillance Requirements for the CREVS will ensure that the control room does analysis assumptions made in support of OL (operating license) Amendment No. 96 are valid. Changes to the control room filtration unit flow rate are more limiting than that currently specified and have already been implemented by resetting the open limit switches on the respective units' outlet dampers. This flow rate is consistent with the design basis for the filtration units as originally licensed.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. Therefore, the possibility of a new or different kind of accident is not created.

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

There will be no margin reduction since these changes are in the conservative direction and have already been approved by NRC via the approval of OL Amendment No. 96. The reduced upper bound flow rate for the control room filtration units is consistent with their design basis and will maintain an average atmosphere residence time greater than or equal to 0.25 sec under both clean and dirty filter conditions. The new charcoal absorber sample laboratory testing protocol is more stringent than the current testing practice and more accurately demonstrates the required performance after a design basis LOCA.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems, necessary to assure the accomplishment of protection functions. There will no impact on the overpower limit, DNBR (departure from nucleate boiling ratio) limits, F_Q, F[delta]H, LOCA PCT (peak cladding temperature), peak local power density, or any other margin of safety. These changes will ensure that the criteria of GDC 19 are met.

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

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: Gail H. Marcus.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Power Plant, Unit Nos. 1 and 2, Town of Two Creeks. Manitowoc County, Wisconsin

Date of amendment request: April 17, 1995.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Section 15.6.2, ''Organization,'' and TS Section 15.6.3, ''Facility Staff Qualifications.'' The training requirements for the Operations Manager and other staff would be changed to provide staffing flexibility.

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

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated; create the possibility of a new or different kind of accident from any previously evaluated; or create the possibility of a new or different kind of accident from any previously evaluated.

1. The proposed change affects only an administrative control, which was based on industry guidance in ANSI N18.1–1971, that recommended the Operations Manager hold an SRO (senior reactor operator) license. This administrative control is being updated to meet the current guidance in ANSI/ANS 3.1–1987.

2. The proposed qualification requirements for the Operations Manager ensures the individual filling the position meets knowledge levels equivalent to the present requirements. It also ensures that individuals responsible for directing the activities of licensed operators continue to hold SRO licenses as required by 10 CFR 50.54(l).

3. Since the proposed specifications ensure regulatory requirements are met and ensures knowledge levels equivalent to existing license requirements for operations management, the proposed changes are considered administrative. The design of plant systems and equipment is not being altered. Plant operations will continue to be directed and performed by qualified personnel. Therefore, the probability or consequences of accidents previously evaluated are not affected, a new or different type of accident is not created, nor is a margin of safety reduced.

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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Gail H. Marcus.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Power Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: April 27, 1995.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Table 15.3.5–1, ''Engineered Safety Features Initiation Instrument Setting Limits,'' and TS Table 15.35–3, ''Engineered Safety Features.'' Setting limits would be modified and references would be changed. The bases for TS Section 15.3.5, ''Instrumentation System,'' would also be changed to be consistent with the TS changes.

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

1. Operation of this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

The probabilities of accidents previously evaluated are based on the probability of initiating events for these accidents.

Initiating events for accidents previously evaluated for Point Beach include: control rod withdrawal and drops, CVCS (chemical and volume control system) malfunction (Boron Dilution), startup of an inactive reactor coolant loop, reduction in feedwater enthalpy, excessive load increase, losses of reactor coolant flow, loss of external electrical load, loss of normal feedwater, loss of all AC power to the auxiliaries, turbine overspeed, fuel handling accidents, accidental releases of water liquid or gas, steam generator tube rupture, steam pipe rupture, control rod ejection, and primary coolant system ruptures.

This license amendment request proposes to correct some minor errors, include appropriate operability requirements for the modification to include the safety injection signal in the time delay for the 4.16KV degraded voltage protection logic, slightly lower the degraded voltage setting limit, change the format of the 4.16 KV degraded voltage and loss of voltage setting limits, and change the time delays associated with the 4.16 KV degraded voltage, 4.16 KV loss of voltage and 480 V loss of voltage protection functions.

These proposed changes do not cause an increase in the probabilities of any accidents previously evaluated because these changes will not cause an increase in the probability of any initiating events for accidents previously evaluated. In particular, these proposed changes affect time delay and format of the setting limits associated with the 4.16 KV degraded voltage, 4.16 KV loss of voltage, and 480 V loss of voltage protection functions. These are protection functions and do not cause accidents.

The consequences of the accidents previously evaluated in the PBNP FSAR (Final Safety Analysis Report) are determined by the results of analyses that are based on initial conditions of the plant, the type of accident, transient response of the plant, and the operation and failure of equipment and systems. The changes proposed in this license amendment request provide appropriate limiting conditions for operation, action settlements, allowable outage times, setting limits, and time delays for the Point Beach Nuclear Plant Technical Specifications for the 4.16 KV degraded voltage, 4.16 KV loss of voltage, and 480 V loss voltage protection functions.

The proposed changes affect functions that are required to ensure the proper operation of engineered features equipment. The proposed changes do not increase the probability of failure of this equipment or its ability to operate as required for the accidents previously evaluated in the PBNP FSAR.

The modifications to reduce the time delay limit associated with the 4.16 KV degraded voltage protection function when the degraded voltage condition is coincident with a safety injection signal, have been designed and installed in accordance with the requirements for PBNP. The probability of occurrence of degraded voltage conditions at PBNP has not been increased. The modifications and proposed Technical Specifications will ensure the proper operation of ESF (engineered safety feature) equipment. These changes do not increase the possibility of failure of this equipment.

Therefore, this proposed license amendment does not affect the consequences of any accident previously evaluated in the Point Beach Nuclear Plant FSAR, because the factors that are used to determine the consequences of accidents are not being changed.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

New or different kinds of accidents can only be created by new or different accident initiators or sequences. New and different types of accidents (different from those that were originally analyzed for Point Beach) have been evaluated and incorporated into the licensing basis for Point Beach Nuclear Plant. Examples of different accidents that have been incorporated into the Point Beach Licensing basis include anticipated transients without scram and station blackout.

The changes proposed by this license amendment request do not create any new or different accident initiators or sequences because these changes to the 4.16 KV degraded voltage, 4.16 KV loss of voltage, and 480 V loss of voltage protection functions will not cause failures of equipment or accident sequences different than the accidents previously evaluated. Therefore, these modifications and proposed Technical Specification changes do not create the possibility of an accident of a different type than any previously evaluated in the Point Beach FSAR.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection.

The changes proposed by this license amendment request provide the appropriate setting limits and time delays for the 4.16 KV degraded voltage, 4.16 KV loss of voltage, and 480 V loss of voltage protection functions. This ensures that the safety systems that protect the reactor and containment will operate as required. The design and operation of the reactor and containment are not affected by these proposed changes. Therefore, the margins of safety for Point Beach are not being reduced because the design and operation of the reactor and containment are not being changed and the safety systems that provide their protection that are being changed are being modified in accordance with the applicable design and installation requirements for Point Beach Nuclear Plant.

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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Gail H. Marcus.

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 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.

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

Date of amendment request: April 21, 1995.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) 3.1.2.4, ''Charging Pumps-Operating,'' by adding a note that indicates that the provisions of TS 3.0.4 and 4.0.4 are not applicable for entry into MODE 4 from MODE 5.

Date of publication individual notice in **Federal Register:** May 2, 1995 (60 FR 21558).

Expiration date of individual notice: June 1, 1995.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, Connecticut 06360.

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 Ch. 1, 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.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: August 19, 1994, as supplemented November 3, 1994.

Brief description of amendment: The amendment requests a line-item improvement to the Radiological Effluent Technical Specifications pursuant to the guidance of Generic Letter 89–01 and incorporates the requirements of revised 10 CFR part 20 and 10 CFR 50.36a.

Date of issuance: May 1, 1995.

Effective date: May 1, 1994.

Amendment No.: 58.

Facility Operating License No. NPF–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** October 12, 1994 (60 FR 51617) The Commission's related evaluation of the amendment, and NRC's response to the public comments received, are contained in a Safety Evaluation dated May 1, 1995.

No significant hazards consideration comments received: Yes.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

Date of application of amendments: November 22, 1994, as supplemented by letters dated January 30, March 2, March 13, and May 2, 1995.

Brief description of amendments: The amendments revise Technical Specification 3.8 to establish restricted loading patterns and associated burnup criteria for placing fuel in the Oconee spent fule pools. In addition, the Design Features sections associated with the reactor and fuel storage are also revised.

Date of issuance: May 3, 1995.

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

Amendment Nos.: 209, 209, and 206.

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

Date of initial notice in **Federal Register:** February 15, 1995 (60 FR 8746); Re-Noticed March 29, 1995 (60 FR 16185).

The May 2, 1995, letter did not change the scope of the November 22, 1994, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 3, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691.

Duquesne Light Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of application for amendment: April 10, 1995, as supplemented April 12, 1995, and April 20, 1995.

Brief description of amendment: This amendment revises Technical Specification 4.6.2.2.d to delete the reference to the specific test acceptance criteria for the Containment Recirculation Spray Pumps and replace the specific test acceptance criteria with reference to the developed head required by the plant's safety analysis. In addition, the 18-month test frequency would be replaced with the test frequency requirements specified in the IST Program. The current footnote (1) pertaining to the performance of recirculation spray pump 2RSS*P21A would be deleted.

Date of issuance: May 3, 1995.

Effective date: May 3, 1995.

Amendment No.: 68.

Facility Operating License No. NPF–73: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (60 FR 19417, April 18, 1995) That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by May 18, 1995, but indicated that if the Commission makes a final no significant hazards consideration any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 3, 1995.

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Unit Nos. 1 and 2, Pope County, Arkansas

Date of amendment request: August 30, 1994 as supplemented January 19, 1995.

Brief description of amendments: The amendments changed requirements related to the site perimeter security system.

Date of issuance: April 28, 1995.

Effective date: April 28, 1995.

Amendment Nos.: Unit 1—Amendment No. 180; Unit 2—Amendment No. 161

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

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18625).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 28, 1995.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: December 14, 1993, as supplemented by letter dated March 3, 1995.

Brief description of amendment: The amendment changed the Appendix A Technical Specifications by removing the reactor vessel material specimen withdrawal schedule and by updating the reactor coolant system pressure-temperature (P–T) curves.

Date of issuance: May 8, 1995.

Effective date: May 8, 1995.

Amendment No.: 106.

Facility Operating License No. NPF–38.: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 19, 1994 (59 FR 2867).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

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

Date of application for amendments: October 20, 1994.

Brief description of amendments: These amendments change the definition of ''core alteration'' to exclude the movement of items not associated with reactivity. The second change involves allowing the personnel airlock (PAL) doors to remain open during fuel movement and core alterations under certain conditions.

Date of issuance: May 11, 1995.

Effective date: May 11, 1995.

Amendment Nos.: 173 and 167.

Facility Operating License No. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55869).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 11, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 28, 1995.

Brief description of amendment: The amendment revises Technical Specification (TS) Section 6.5.1.12 to delete the requirement to render determinations in writing with regard to whether or not activities listed in TS Sections 6.5.1.2 and 6.5.1.5 constitute an unreviewed safety question. These activities are changes to Appendix A Technical Specifications (6.5.1.2) and investigations of all violations of the TSs (6.5.1.5). This change is consistent with NUREG–1433 Standard Technical Specifications General Electric Plants, BWR/4 Revision 0, dated September 28, 1992.

Date of issuance: May 1, 1995.

Effective date: May 1, 1995.

Amendment No.: 180.

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

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16188).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 1, 1995.

No significant hazards consideration comments received: Yes.

By letter dated April 5, 1995, Mr. Kent W. Tosch, of the State of New Jersey Department of Environmental Protection commented that they concur with GPU Nuclear's rationale that these unreviewed safety question reviews serve no value since these activities specifically require NRC review and approval. The State official had no other comments.

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Dos. 50–498 and 50–499, South Texas Projects, Units 1 and 2, Matagorda County, Texas

Date of amendment request: February 15, 1995.

Brief description of amendment: The amendment modified Technical Specification 4.6.2.3.a.2 (and associated Bases) to reflect the reactor containment fan cooler flow rate assumed in the accident analysis and to specify that this flow is provided by the component cooling water system.

Date of issuance: May 2, 1995.

Effective date: May 2, 1995, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 74; Unit 2—Amendment No. 63.

Facility Operating License Nos. NPF–76 AND NPF–80. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16189) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 2, 1995.

No significant hazards consideration comments received: No.

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

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50–461, Clinton Power Station, Unit No. 1. DeWitt County, Illinois

Date of application for amendment: February 10, 1995.

Brief description of amendment: The amendment changes Technical Specification 3.3.2.1, ''Control Rod Block Instrumentation,'' to revise two surveillance requirements and their associated notes for the Rod Withdrawal Limiter mode of the Rod Pattern Control System. The changes are consistent with the Clinton Power Station Technical Specifications prior to implementation of the improved Technical Specifications (Amendment No. 95) and eliminates the potential for unnecessary power reductions.

Date of issuance: May 2, 1995.

Effective date: May 2, 1995.

Amendment No.: 100.

Facility Operating License No. NPF–62. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register: March 29, 1995. (60 FR 16190)**

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 1995.

No significant hazard consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727.

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

Date of application for amendment: July 22, 1993, as supplemented February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995.

Brief description of amendment: The amendment modified Facility Operating License No. NPF–69 and the NMP–2 TSs to authorize an increase in the maximum power level of NMP–2 from 3323 megawatts thermal (MW_t) to 3467 MW_t. The amendment also approves changes to the TSs to implement uprated power operation.

Date of issuance: April 28, 1995.

Effective date: As of the date of issuance to be implemented prior to restart from refueling outage number 4.

Amendment No.: 66.

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

Date of initial notice in **Federal Register: March 16, 1994 (59 FR 12360). The letters dated February 4, August 23, September 16, October 6, and December 2, 1994, and January 3, January 9, March 8, and April 10, 1995, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.**

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 28, 1995.

No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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: October 18, 1994, a supplemented February 21, 1995.

Brief description of amendment: The amendment changes Surveillance Requirement 4.6.1.2.a (Overall Integrated Containment Leakage Rate Tests) by revising the surveillance interval for Type A tests from 40 plus or minus 10 months to approximately equal intervals during each 10-year inservice period. The amendment also removes a note that expired upon completion of Cycle II refueling outage.

Date of issuance: May 3, 1995.

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

Amendment No.: 187.

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

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16191).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated may 3, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London turnpike, Norwich, CT 06360.

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: December 23, 1994.

Brief description of amendment: The amendment changes the acceptance criteria for the peak transient generator voltage from 4784 volts to 5000 volts during full load rejection tests of the diesel generator (DG), and also deletes the 10-year surveillance requirement to perform a 110% pressure test of the DG fuel oil system.

Date of issuance: May 1, 1995.

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

Amendment No.: 110.

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

Date of initial notice in **Federal Register:** February 15, 1995 (60 FR 8751).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 1, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

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: September 28, 1994.

Brief description of amendment: The amendment revises Surveillance Requirement 4.6.1.2.a of the Technical Specification to eliminate the requirement to perform Type A tests on an interval of 40 plus or minus 10 months while reiterating the Appendix J requirement that the Type A tests be performed three times, at approximately equal intervals, during each 10 year service period. In addition, a footnote is added which states that the third Type A test will be performed during the sixth refueling outage. This reflects an exemption to Appendix J which separates the third Type A test from the 10 year inservice inspection.

Date of issuance: May 8, 1995.

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

Amendment No.: 111.

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

Date of initial notice in **Federal Register:** November 23, 1994 (59 FR 60384)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1995.

No significant hazards consideration comments received: NO.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

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

Date of application for amendment: August 19, 1994, as supplemented March 15, 1995.

Brief description of amendment: The amendments add a new action statement to Technical Specification 3.1.3.2.1., ''Position Indication Systems—Operating''.

Date of issuance: May 3, 1995.

Effective date: May 3, 1995.

Amendment No.: 166 and 148.

Facility Operating License Nos. DPR–70 and DPR–75. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 12, 1994 (59 FR 51626) The March 15, 1995 supplement 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 3, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 19, 1993; superseded May 16, 1994; superseded February 10, 1995; supplemented February 17, 1995 (TS 93–04).

Brief description of amendment: The amendments clarify the Limiting

Conditions for Operation applicable to the dual function of the containment vacuum relief isolation lines by specifying the actions that would be required should one or more of the vacuum relief isolation lines by specifying the actions that would be required should one or more of the vacuum relief lines be incapable of performing the containment isolation function or incapable of performing the vacuum relief function.

Date of issuance: April 28, 1995.

Effective date: April 28, 1995.

Amendment No.: 197 and 188.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** May 12, 1994 (58 FR 28060); renoticed June 22, 1994 (59 FR 32237), and March 29, 1995 (60 FR 16202).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 28, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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

Date of application for amendment: November 15, 1994; superseded March 7, 1995 (TS 94–12).

Brief description of amendments: The amendments remove the frequencies specified in the Technical Specifications for performing audits and delete the requirement to perform the Radiological Emergency Plan, Physical Security Plan, and Safeguard Contingency Plan reviews.

Date of issuance: May 10, 1995.

Effective date: May 10, 1995.

Amendment No.: 198 and 189.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65823); renoticed March 29, 1995 (60 FR 16203)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 10, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of application for amendment: January 30, 1995.

Brief description of amendment: This amendment revises Technical Specification (TS) 4.6.1.2.a, ''Containment Systems, Containment Leakage, Surveillance Requirements (SR)'' and Bases 3/4.6, ''Containment Systems,'' to state that Type A tests for overall integrated containment leakage rate testing shall be conducted in accordance with the requirements specified in appendix J of 10 CFR part 50, as modified by NRC-approved exemptions. Additionally, TS SR 4.6.1.2.a.

Date of issuance: May 3, 1995.

Effective date: May 3, 1995.

Amendment No.: 198.

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

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14028).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

Date of application for amendments: July 8, 1993, as supplemented by letters dated July 12, 1994, and March 7, 1995.

Brief description of amendments: The amendments revise the NA–1&2 Technical Specifications by deleting the requirements to periodically review certain administrative and technical procedures.

Date of issuance: May 1, 1995.

Effective date: May 1, 1995.

Amendment Nos.: 190 and 171.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 4, 1993 (58 FR 41518).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 1, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 27, 1993, as supplemented September 6, 1994, and March 7, 1995.

Brief description of amendments: The amendments revise the NA–1&2 Technical Specifications regarding the review responsibilities of the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee.

Date of issuance: May 2, 1995.

Effective date: May 2, 1995.

Amendment Nos.: 191 and 172.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 16, 1994 (59 FR 7700).

The September 6, 1994, and March 7, 1995 submittals provided additional information only, and did not change the staff's initial proposed determination of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 2, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 28, 1991.

Brief description of amendments: These amendments incorporate operability and surveillance requirements for power-operated relief valves to conform with Generic Letter 90–06.

Date of issuance: May 2, 1995.

Effective date: May 2, 1995.

Amendment Nos.: 198 and 198.

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

Date of initial notice in **Federal Register:** October 2, 1991 (56 FR 49929).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By June 23, 1995, 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 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 lest one contention will not be permitted to participate as a party.

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (*Project Director*): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

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

Date of application for amendments: April 24, 1995.

Brief description of amendments: the amendments change the Technical Specifications by modifying the surveillance testing periodicity requirements of the automatic actuation logic of engineered safeguards equipment.

Date of issuance: May 5, 1995.

Effective date: May 5, 1995.

Amendment Nos.: 162 and 150.

Facility Operating Licenses Nos. DPR–39 and DPR–48. The amendments revised the Technical Specifications.

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

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 5, 1995.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690.

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

NRC Project Director: Robert A. Capra.

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Pant, Unit No. 1, Calvert County, Maryland

Date of application for amendment: April 28, 1995.

Brief description of amendment: The amendment revises the control room emergency ventilation system TS 3.7.6.1, Limiting Condition For Operation. The revision extends the one-time increase in the allowed outage time for loss of emergency power only, from the 30 days previously approved, to 45 days. This extension is necessary to allow time to repair the Number 21 emergency diesel generator which failed its operability tests subsequent to modifications which have been recently completed.

Date of issuance: May 2, 1995.

Effective date: As of the date of issuance to be implemented upon receipt.

Amendment No.: 205.

Facility Operating License No. DPR–53: Amendment revised the Technical Specifications.

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

The Commission's related evaluation of the amendment, consultation with the State, and final determination of no significant hazards consideration are continued in a Safety Evaluation dated May 2, 1995.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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

NRC Project Director: Ledyard B. Marsh.

Dated at Rockville, MD, this 17th day of May, 1995.

For the Nuclear Regulatory Commission,

Elinor G. Adensam,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–12538 Filed 5–22–95; 8:45 am]

BILLING CODE 7590–01–M

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–35723; File No. SR–Amex–95–08]

Self-Regulatory Organizations; American Stock Exchange, Inc.; Order Granting Approval to Proposed Rule Change Relating to Membership Structure and Requirements and the Exchange's Gratuity Fund

May 16, 1995.

I. Introduction

On February 17, 1995, the American Stock Exchange, Inc. (''Amex'' or ''Exchange'') filed with the Securities and Exchange Commission (''SEC'' or ''Commission''), pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] a proposed rule change to amend its Construction, Rules and Membership Lease Plan to allow organizations, including certain pension plans, to own memberships legally as well as beneficially and to allow individuals and organizations to own multiple memberships. The Exchange also is proposing to revise its Gratuity Fund to reflect the above changes, to increase the death benefit paid thereunder, and to allow options principal members to participate therein.

The proposed rule change was published for comment in Securities

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.