Week of December 9—Tentative

Thursday, December 12

3:30 p.m.
Affirmation Session (Public Meeting)

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (Recording—(301) 415–1292).

CONTACT PERSON FOR MORE INFORMATION: Bill Hill (301) 415–1661.

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

This notice is distributed by mail to several hundred subscribers: if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661).

In addition, distribution of this meeting notice over the internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: November 15, 1996.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 96–29680 Filed 11–15–96; 2:39 pm]

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NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 28, 1996, through November 7, 1996. The last biweekly notice was published on November 6, 1996.

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

By December 20, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: 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 20555-0001, and to the attorney for the licensee.

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

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

Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: October 2, 1996

Description of amendment request: The amendment would change Figures 3.1.A-1, 3.1.A-2, and 3.1.A-3, Section 3.1.B and its Bases, Figures 3.1.B-1 and 3.1.B-2, and the Bases of Section 4.3 and Figure 4.3-1 of the Technical Specifications by providing new pressure/temperature limit curves.

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

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

Response:

Neither the probability nor the consequences of an accident previously analyzed is increased due to the proposed changes. The adjusted reference temperature of the most limiting beltline material was used to correct the pressure-temperature (P-T) curves to account for irradiation effects. Thus, the operating limits are adjusted to incorporate both the initial fracture toughness conservatism present when the reactor vessel was new and the effect of fluence. The adjusted reference temperature calculations were performed utilizing the guidance contained in RG [Regulatory Guide] 1.99, Revision 2. Overpressure Protection System (OPS) curves and tables were regenerated to be consistent with the new P-T curves. The updated curves provide assurance that brittle fracture of the reactor vessel is prevented.

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

Response:

The updated P-T and OPS limits will not create the possibility of a new or different kind of accident. The revised operating limits merely update the existing limits by taking into account the effects of radiation embrittlement, utilizing criteria defined in RG 1.99, Revision 2. The updated curves are conservatively adjusted to account for the effect of irradiation on the limiting reactor vessel material.

No change is being made to the way the pressure-temperature limits provide plant protection. No new modes of operation are involved. Incorporating this amendment does not necessitate physical alteration of the plant.

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

Response:

The proposed amendment does not involve a significant reduction in the margin of safety. The pressure-temperature operating limits and OPS setpoints are designed to maintain an appropriate margin of safety. The required margin is specified in ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code, Section III, Appendix G and 10 CFR [Part] 50 Appendix G. The revised curves are based on the latest NRC guidelines along with actual neutron fluence data for the reactor vessel. The new limits retain a margin of safety equivalent to the original margin when the vessel was new and the fracture toughness was slightly greater. The new operating limits account for irradiation embrittlement effects, thereby maintaining a conservative margin of safety.

The removal of the pressure-temperature limits for criticality does not reduce the plant safety margin because these limits are conservatively encompassed and bounded by the requirements of the proposed Technical Specification 3.1.C.2.

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

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Project Director: S. Singh Bajwa, Acting

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: September 6, 1996

Description of amendment request: The proposed amendments would revise Item 7.c of BVPS-1 Technical Specifications (TSs) Table 3.3-3 and Item 7.d of BVPS-2 TS Table 3.3-3 to reflect that a safety injection (SI) signal starts all auxiliary feedwater (AFW) pumps. The notation on BVPS-1 TS Table 3.3-5 would be revised to state that the response time is for all AFW pumps on all SI signal starts. Items 7.d of BVPS-2 TS Tables 3.3-4 and 4.3-2 would be revised to reflect that an SI signal starts all AFW pumps.

The proposed amendments would also revise and reformat TSs 3/4.7.1.2 to more closely resemble the wording contained in the NRC's ''Standard Technical Specifications Westinghouse Plants,'' (NUREG-1431, Revision 1). These changes would require three AFW trains to be operable and would provide what constitutes an operable train. The mode applicability for these TSs would expand to include Mode 4 when the steam generator(s) is relied upon for heat removal.

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

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

The proposed revisions to reflect that a Safety Injection (SI) signal starts the turbine driven Auxiliary Feedwater (AFW) pump, in addition to both motor driven AFW pumps, will ensure that plant operability requirements for the AFW system actuation signals are maintained at a level consistent with current safety analyses. The proposed revisions to Limiting Condition for Operation (LCO) 3.7.1.2 will require that the AFW pumps and associated flow paths are maintained operable to ensure that the AFW system can mitigate the consequences of a Design Basis Accident (DBA) with a loss of normal feedwater. The addition of the Mode 4 applicability will ensure that a safety related source of cooling water is available to remove decay heat.

The proposed change will ensure that the plant is placed in Mode 4 when the number of operable feedwater injection headers is insufficient to ensure that at least two steam generators are supplied during a feedline break accident.

The proposed addition of footnote (2) to action statement ''c'' will limit plant thermal cycles following a refueling outage due to turbine driven AFW pump inoperability. During the additional time period provided by footnote (2) to reach Hot Shutdown, the two remaining motor driven AFW pumps will provide sufficient flow to the steam generators to mitigate the consequences of a DBA assuming no single failures during this time period. Since there is negligible decay heat following a refueling outage prior to entry into Mode 2, the performance capabilities of the two remaining motor driven AFW pumps to remove decay heat will not be challenged.

Changing the AFW pump surveillance test frequencies for Beaver Valley Power Station (BVPS) Unit No. 2 to quarterly, as specified in the Inservice Testing (IST) Program, will continue to assure that the AFW system will be capable of performing its intended functions.

The proposed change to the current Surveillance Requirement 4.7.1.2, for BVPS Unit No. 2 only, will not lower the pump performance operability criteria for the AFW pumps. The required values for developed pump head and flow will continue to satisfy accident mitigation requirements and will be maintained and controlled in the BVPS Unit No. 2 IST Program. Future changes to the AFW pump head and flow requirements will be made under the 10 CFR 50.59 process to ensure that the AFW design requirement to remove sufficient decay heat continues to be met.

Based on the above factors, the probability of an accident previously evaluated is not significantly increased.

The proposed changes do not affect the ability of the AFW system to perform as assumed in the safety analyses. The proposed changes will not result in any additional challenges to plant equipment. Because the plant design limits will continue to be met, the fuel and reactor coolant system pressure boundary integrity is not challenged for the assumptions employed in the calculation of the offsite radiological doses. The additional time to reach Mode 4 from Mode 3 provided by footnote (2) does not result in increased radiological consequences. The potential for a radioactivity release due to the uncontrolled heatup of [the] reactor coolant system[s] are enveloped by the releases postulated in the DBA Loss of Coolant Accident (LOCA) analysis in the Updated Final Safety Analysis Report. The DBA LOCA analysis assumes 102% power operation prior to the event and assumes that core melt occurs. Therefore, there is no increase in the radiological consequences as a result of allowing additional time to repair/test the turbine driven AFW pump. Hence, the consequences of a DBA previously evaluated is not significantly increased.

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

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

The proposed change does not alter the method of operating the plant. The AFW system is an accident mitigation system and is normally in standby. System operation is initiated in response to a DBA. The AFW pumps will continue to provide sufficient flow to mitigate the consequences of a DBA. AFW operation continues to fulfill the safety function for which it was designed and no changes to plant equipment will occur. As a result, an accident which is new or different than any already evaluated in the Updated Final Safety Analysis Report will not be created due to this change.

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

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

The proposed changes will not affect the heat removal capability of the AFW system to a value less than assumed in the safety analysis. The proposed changes will not result in any additional challenges to the plant equipment including the fuel and reactor coolant system pressure boundary. The additional time period to reach Hot Shutdown provided by footnote (2) will not significantly reduce the decay heat removal capability provided by the AFW system. The two remaining motor driven AFW pumps will continue to provide sufficient flow to the steam generators as assumed in the safety analysis to mitigate the consequences of a DBA assuming no single failure during this time period. The plant will continue to operate within the bounds of the safety analysis.

The AFW system will continue to be tested in a manner and at a frequency which will ensure acceptable system performance should it be relied upon to remove decay heat following a DBA.

The AFW pumps' performance requirements will continue to be controlled in a manner to ensure safety analysis assumptions are met.

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

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: July 25, 1996

Description of amendment request: The proposed change modifies Technical Specification (TS) 3/4.7.4 Ultimate Heat Sink (UHS) by incorporating more restrictive fan operability requirements and lower basin temperature. Several other administrative changes are incorporated to improve the humanfactors associated with this TS.

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

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

Response: No

The proposed change modifies the UHS TS by revising [Wet Cooling Tower] WCT basin water temperature from less than or equal to 95 Degrees Fahrenheit to less than or equal to 89 Degrees Fahrenheit and incorporating more restrictive cooling tower fan operability requirements. These changes are necessary to adequately preserve the assumptions and limits of the revised UHS design basis calculations. These calculations conclude that the UHS is capable of dissipating the maximum peak heat load resulting from the limiting design bases accident (i.e., large break LOCA) and the most severe natural phenomena (i.e., tornado event). Other changes are purely administrative in nature. The proposed change does not directly affect any material condition of the plant that could directly contribute to causing an accident. The proposed change ensures that the mitigating effects of the UHS will be consistent with the design basis analysis. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Response: No

The proposed change modifies the UHS TS to be consistent with revised design basis calculations. These new calculations adjust margin to incorporate an additional allowance for fouling in the [Component Cooling Water] CCW heat exchangers and more restrictive UHS minimum fan requirements that were not adequately addressed in the initial design basis. This change also incorporates administrative changes that are intended to improve the application and use of this specification. The proposed change will not alter the operation of the plant or the manner in which the plant is operated. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed change modifies the UHS TS by revising WCT basin water temperature from less than or equal to 95 Degrees Fahrenheit to less than or equal to 89 Degrees Fahrenheit and incorporating more restrictive cooling tower fan operability requirements. Modifying the UHS meteorological design bases reduced WCT basin temperature requirement for operability, thus, providing an allowance for fouling in the CCW heat exchangers. The proposed change better preserves the margin of safety by ensuring that the UHS will maintain the CCW accident analysis temperature limit of 115 Degrees Fahrenheit. Increased cooling tower fan operability requirements will ensure that the expected cooling efficiency is actually available and not unknowingly degraded due to fouling. Other changes requested herein are purely administrative in nature, do no affect safety margins and intended to improve the use and application of this specification. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: October 4, 1996

Description of amendment request: The proposed amendments would incorporate the requirements necessary to change the basis for prevention of criticality in the fuel storage pool. This change would eliminate credit for Boraflex as a neutron absorbing material in the fuel storage pool criticality analysis and would support the storage of fuel with enrichments up to and including 5.0 weight percent U-235 rather than the current value of 4.5 weight percent U-235.

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.

There is no increase in the radiological consequences of accidents previously evaluated in the Vogtle FSAR [Final Safety Analysis Report] with the use of 5.0 weight percent U-235 fuel. Increasing the enrichment up to and including 5.0 weight percent U-235 affects the radiological source terms and subsequently the potential releases both normal and accidental. Evaluations performed (WCAP-12610-P-A, Reference 6) considered the source term, gap fraction, normal operating plant releases and the accident doses for a maximum fuel enrichment of 5.0 weight percent U-235. It was concluded that operating with and storing fuel with 5.0 weight percent U-235 enrichment may result in minor increases in the normal annual releases of long half-life fission products that are not significant. Also, the radiological consequences of accidents are minimally affected due to the very small changes in the core inventory and the fact that the currently assumed gap fractions remain bounding.

The use of the slightly higher enrichment for VEGP [Vogtle Electric Generating Plant] fuel will not result in burnups in excess of those currently allowed for VEGP. The cycle design methods and limits will remain the same as are currently licensed. Therefore the use of fuel with the higher enrichment is not expected to result in operating conditions outside those currently allowed for VEGP.

There is no increase in the probability of a fuel assembly drop accident in the fuel storage pool when considering the presence of soluble boron in the pool water for criticality control. The handling of the fuel assemblies in the fuel storage pool has always been performed in borated water.

Fuel assembly placement will be controlled pursuant to approved fuel handling procedures and will be in accordance with the spent fuel rack storage configuration limitations in the COLR [Core Operating Limit Report]. The consequences of a misplaced assembly have been included in the analysis supporting this revision to the Technical Specifications.

There is no increase in the consequences of the accidental misloading of a spent fuel assembly into the fuel storage pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading of an assembly even considering a dilution event. The proposed Technical Specifications and COLR limitations will ensure that an adequate fuel storage pool boron concentration will be maintained.

There is no increase in the probability of the loss of normal cooling to the fuel storage pool water due to the presence of soluble boron in the pool water for subcriticality control, because a high concentration of soluble boron has been maintained in the fuel storage pool water.

The loss of normal cooling to the fuel storage pool will cause an increase in the temperature of the fuel storage pool water. This will cause a decrease in water density which would normally result in an addition of negative reactivity. However, since Boraflex is not considered to be present, and the fuel storage pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. The amount of soluble boron required to offset this postulated accident was evaluated for the allowed storage configurations. The amount of soluble boron necessary to mitigate these accidents and ensure that the K_{eff} will be maintained less than or equal to 0.95 has been included in the fuel storage pool boron concentration. Because adequate soluble boron will be maintained in the pool water, the consequences of a loss of normal cooling to the fuel storage pool will not be increased.

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

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

The potential for criticality accidents in the fuel storage pool are not new or different types of accidents. It has been reanalyzed in the Criticality Analysis report (Enclosure 5 [of the proposed amendment request]).

Because soluble boron has been maintained in the fuel storage pool water, the possibility of a fuel storage pool dilution has previously existed. Therefore, the implementation of Technical Specification controls for the soluble boron will not create the possibility of a new or different kind of accidental pool dilution.

With credit for soluble boron now a major factor in controlling criticality, an evaluation of fuel storage pool dilution events was completed. A generic methodology was applied... to identify potential events which would dilute the soluble boron contained in PWR [pressurized water reactor] fuel storage pools, and to quantify the frequency of those events. This methodology utilized a probabilistic assessment of a composite plant model to calculate the event frequency of a dilution event. The results of the assessment concluded that the event frequency remained less than the NRC Safety Goal Policy Statement target risk objective of IE-6/reactor year.

Differences between the composite plant described in WCAP-14181 and Vogtle relative to the potential sources of pool dilution were addressed in an individual analysis of the Vogtle pool. This analysis was conducted with methodology which closely paralleled that employed in WCAP-14181. That analysis, found in Enclosure 6 [of the licensee's proposed amendment request], concluded that the frequency of pool dilution to the 0.95 K_{eff} boron concentration (1250 ppm) is on the same order of magnitude as reported in WCAP-14181 and less than the NRC Safety Goal Policy Statement criterion of 1.0E-6/reactor year.

Proposed Technical Specifications 3.7.17 and 3.7.18 which ensure the maintenance of the fuel storage pool boron concentration and storage configuration, do not represent new concepts. The actual boron concentration in the fuel storage pool has been maintained at a higher value than the proposed limits for the Unit 1 and 2 fuel storage pools for refueling purposes. The criticality analysis (Enclosure 5 [of the licensee's proposed amendment request]) determined that a boron concentration of 1,100 ppm (Unit 1) and, 1,250 ppm (Unit 2) results in a $K_{eff}<0.95$ including all the calculational uncertainties and additional margin to compensate for the possibility of loss of cooling, or a misplaced assembly.

There is no significant change in plant configuration, equipment design, or usage of plant equipment. The safety analysis for dilution accidents has been expanded; however, the criticality analyses assure that the pool will remain subcritical with no credit for soluble boron. Therefore, the proposed changes will not create the possibility of a new or different kind of accident.

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

Proposed Technical Specifications 3.7.17 and 3.7.18 and the associated spent fuel boron concentration and storage limits in the COLR will provide adequate safety margin to assure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis (Enclosure 5 [of the licensee's proposed amendment request]) performed in accordance with the Westinghouse criticality analysis methodology...

While the cricality analysis utilized credit for soluble boron, a storage configuration has been defined using maximum feasible K_{eff} calculations to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron under normal storage conditions and assuming nominal fuel assembly parameters and fuel rack dimensions. Soluble boron credit is used to offset uncertainties, tolerances and off-normal conditions (such as a misplaced assembly) and to provide subcritical margin such that the fuel storage pool K_{eff} is maintained less than or equal to 0.95.

The loss of a considerable amount of soluble boron in the fuel storage pool which could lead to exceeding a K_{eff} of 0.95 during accidents and under adverse conditions has been evaluated and shown to be very improbable.

The combination of the probabilistic evaluation which shows that the dilution of the fuel storage pool is a low probability occurrence, the maximum feasible K_{eff} calculation which shows that the K_{eff} will remain less than 1.0 when flooded with unborated water and assuming nominal fuel assembly parameters and fuel rack dimensions, and the unavailability of the large volumes of water which are necessary to dilute the fuel storage pool, provide a level of safety comparable to the conservative criticality analysis methodology...

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's 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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308

NRC Project Director: Herbert N. Berkow

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: September 25, 1996

Description of amendment request: The proposed amendment would (1) revise the required number of operable gaseous radioactivity monitoring system channels and particulate radioactivity monitoring system channels from one in each of the monitoring systems to one in either of the monitoring systems, (2) allow both the gaseous radioactivity monitoring system and the particulate monitoring system to be inoperable for up to 30 days provided that grab samples are obtained and analyzed at least once per 12 hours, and (3) add an action for the loss of all reactor coolant system leakage detection systems (drywell floor sump level monitoring system, gaseous radioactivity monitoring system and particulate radioactivity monitoring system).

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

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

The function of the reactor coolant system leakage detection systems is to detect leakage from the reactor coolant pressure boundary so that appropriate actions can be taken before the integrity of the reactor coolant pressure boundary is impaired. In the plant accident analysis, no credit for mitigation of an accident is taken for the reactor coolant system leakage detection systems. These proposed changes do not alter this function, therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The function of the reactor coolant system leakage detection systems is to detect leakage from the reactor coolant pressure boundary so that appropriate actions can be taken before the integrity of the reactor coolant pressure boundary is impaired. These proposed changes do not alter this function; therefore, these changes do not create the possibility of a new or different kind of accident previously evaluated.

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

The change to allow both the gaseous and particulate radioactivity monitoring systems to be inoperable at the same time provided a grab sample is obtained and analyzed at least once per 12 hours is predicated on the availability of the primary leak detection system (drywell floor sump level monitor system). Since the gaseous and particulate radioactivity monitoring systems are backups to the drywell floor sump level monitoring system, allowing grab samples every 12 hours provides periodic information that is adequate to detect leakage. The addition of the action to require an orderly shutdown of the unit for the loss of all reactor coolant system leakage detection systems does not affect the margin of safety. Therefore, these proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701

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

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: October 24, 1996

Description of amendment request: The proposed amendments would change Technical Specification 3/4.7.1.2, ''Auxiliary Feedwater System.'' The changes would revise the 18-month surveillances performed on the system's pumps and valves because testing of the turbine driven Auxiliary Feedwater pump (TDAFWP) can only be performed in higher modes when there is sufficient secondary steam pressure.

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 changes proposed on the testing of components in the AFW [Auxiliary Feedwater] System do not affect the operation of the equipment during conditions when they are required to perform their safety function. No physical changes to the plant result from the proposed changes made to the surveillance requirements. The AFW System is used as a backup system upon loss of main feedwater which is analyzed as a Condition II event in the UFSAR [Updated Final Safety Analysis Report] and as such, does not impact the probability of an accident.

Testing is being performed with the plant in the condition in which the automatic initiation signals would result, that is, with the plant in Hot Standby. The changes do not impact the availability of the AFW System in providing feedwater to the steam generators. The 24 hour duration to perform testing is sufficiently short that it is considered unlikely that a condition requiring AFW initiation would occur with the TDAFWP unable to feed the generators. For such an occurrence, however, the motor driven AFW pumps would be available to mitigate the consequences of the event. This time is less than the 72 hour allowed outage time for an inoperable TDAFWP in Modes 1-3.

Therefore, the consequences of an accident previously evaluated are not significantly increased.

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

The proposed changes do not involve any modifications to existing plant equipment, do not alter the function of any plant systems, do not introduce any new operating configurations or new modes of plant operation, nor change the safety analyses. Testing of the TDAFWP in Mode 3, Hot Standby, will not impact auxiliary feedwater capability or impact the ability to maintain Reactor Coolant temperature. The proposed changes will, therefore, not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The changes to the valve surveillance does not decrease the scope of the existing testing, but will clarify the automatic valves to be included.

The time in which testing is performed, within 24 hours of reaching 680 psig steam generator pressure, ensures that testing is performed in a timely manner after attaining the required steam pressure. This does not impose a significant safety impact since the testing is performed within the plant at the zero load conditions prior to increasing reactor power.

Elimination of the wording ''during shutdown,'' in reference to the time in which the surveillance is performed, is considered editorial and is proposed for consistency with the change made to the pump surveillance requirement.

All changes are consistent with the intent of Salem's current TS and with the 18 month surveillances specified in NUREG-1431, Revision 1.

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

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: September 30, 1996 (TSCR 192)

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Section 15.3.3, ''Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray,'' TS 15.3.7, ''Auxiliary Electrical Systems,'' and the TS Bases to reflect proposed changes to the limiting conditions for operation, action statements, allowable outage times, and design specifications for the Point Beach Nuclear Plant (PBNP) TS associated with the containment accident fan coolers, service water equipment, and normal and emergency power supplies.

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 drop, CVCS [chemical volume and 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 waste liquid or gas, steam generator tube rupture, steam pipe rupture, control rod ejection, and primary coolant system ruptures.

This license amendment request proposes to change the limiting conditions for operation, action statements, allowable outage times, and design specifications for the Point Beach Nuclear Plant Technical Specifications associated with the containment accident fan coolers, service water equipment, and normal and emergency power supplies.

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 changes affect accident mitigation systems and equipment which 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 statements, and allowable outage times for service water, containment cooling and normal and emergency power supplies.

The proposed changes affect components that are required to ensure the proper operation of engineered safety 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 proposed changes that increase the allowed outage times for engineered safety features equipment continue to provide appropriate limitations for these conditions because sufficient equipment is still required to be operable for accident mitigation and the proposed allowed outage times are consistent with currently accepted time periods for these situations.

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 limiting conditions for operation, action statements, allowable outage times, and design specifications for service water, containment cooling and normal and emergency power supplies will not cause failures of equipment or accident sequences different than the accidents previously evaluated. Therefore, these 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 limiting conditions for operation, action statements, allowable outage times, and design specifications for service water, containment cooling and normal and emergency power supplies. This ensure 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 and limiting conditions of operation for these safety systems that provide their protection that are being changed will continue to meet the requirements for accident mitigation 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: John N. Hannon

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

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

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

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

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

Carolina Power & Light Company, et al., Docket Nos. 50-325 & 50-324, Brunswick Steam Electric Plant, Units 1 & 2, Brunswick County, North Carolina

Date of amendment request: April 2, 1996 (BSEP 96-0123), as supplemented by an earlier submittal dated November 20, 1995 (BSEP 95-0535), and by subsequent submittals dated July 1, 1996 (BSEP 96-0242), July 30, 1996 (BSEP 96-0287), August 7, 1996 (BSEP 96-0300), September 13, 1996 (BSEP 96-0340), September 20, 1996 (BSEP 96-0348), October 1, 1996 (BSEP 96-0362), October 22, 1996 (BSEP 96-0392), October 22, 1996 (BSEP 96-0403), and October 29, 1996 (BSEP 96-0412).

Brief description of amendment: The proposed amendment would modify Facility Operating Licenses Nos. DPR-71 and DPR-62 and the Technical Specifications (TS) for the Brunswick Steam Electric Plant, Units 1 and 2, respectively, to authorize an increase in the maximum power level from 2436 megawatts thermal (MWt) to 2558 MWt.

Date of issuance: November 1, 1996

Effective date: November 1, 1996

Amendment No.: 183 (Unit 1); 214 (Unit 2)

Facility Operating License Nos. DPR-71 and DPR-62: Amendment revises

Facility Operating License Nos. DPR-71 and DPR-62 and the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25698) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 1, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: August 5, 1994, as supplemented by letters dated November 17, 1994, December 2, 1994, and August 1, 1996.

Brief description of amendment: The amendment revises surveillance intervals for various systems, components and instruments to accommodate a 24-month refueling cycle. These revisions are being made in accordance with the guidance provided by Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.''

Date of issuance: October 30, 1996

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

Amendment No.: 187

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 7, 1994 (59 FR 63117) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: December 11, 1995, as supplemented by letters dated January 15, September 3, October 2, October 18, and October 25, 1996.

Brief description of amendment: The amendment revises the Administrative Controls section of the TS by deleting or relocating requirements that are adequately controlled by existing regulatory requirements, adding requirements, and editorially restructuring the TS to be consistent with NUREG-1432, ''Standard Technical Specifications, Combustion Engineering Plants.'' In addition, containment leak rate testing requirements are revised to allow the Type A integrated leak rate test to be scheduled in accordance with Option B of 10 CFR Part 50, Appendix J. Review of several changes proposed by the licensee have not yet been completed by the staff. The NRC will issue an evaluation of these changes upon completion of staff review.

Date of issuance: October 31, 1996

Effective date: October 31, 1996

Amendment No.: 174

Facility Operating License No. DPR-20 Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 20, 1996 (61 FR 49493). The October 2, October 18, and October 25, 1996, letters provided clarifying information and updated TS pages that were within the scope of the initial application and did not affect the staff's initial proposed no significant hazards consideration determination. Therefore, renoticing was not warranted.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31,1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, IllinoisDate of application for amendments: April 8, 1996, as supplemented on October 14, 1996.

Brief description of amendments: The amendments revise various sections of the Technical Specifications (TS) to reflect the transition of fuel supplier from General Electric (GE) to Siemens Power Corporation (SPC). The amendments revise the definitions, limiting conditions for operation, required actions, or surveillance requirements related to the following fuel thermal limits: Linear Heat Generation Rate, Critical Power Ratio, Minimum Critical Power Ratio, and Average Planar Linear Heat Generation Rate. The previous GE terminology is replaced with vendor independent terms and new, NRC-approved methodologies are incorporated. The amendments also include changes to Section 6.0 of the TS to include SPC references, relocate the requirements for the traversing in-core probe system from the TS to the Core Operating Limits Report, and revise the fuel description in TS Section 5.0.

Date of issuance: October 29, 1996

Effective date: Immediately, to be implemented prior to startup of Cycle 9 for Unit 1 and Cycle 8 for Unit 2.

Amendment Nos.: 116, 101

Facility Operating License Nos. NPF-11 and NPF-18: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25699) The October 14, 1996, submittal provided additional 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 October 29, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, IllinoisDate of application for amendments: August 16, 1996, as supplemented on October 4, 1996.

Brief description of amendments: The amendments revise the definition of the F* distance by removing the uncertainty term from the specified distance and removing the footnote which specifies the time frame for which it is applicable.

Date of issuance: November 6, 1996

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 174, 161

Facility Operating License Nos. DPR-39 and DPR-48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47968) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 6, 1996No significant hazards consideration comments received: No

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

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

Date of application for amendments: September 3, 1996

Brief description of amendments: The amendments incorporate revised installation procedures for steam generator tube sleeves designed by ABB Combustion Engineering (ABB/CE).

Date of issuance: October 29, 1996

Effective date: October 29, 1996

Amendment Nos.: 173 and 160

Facility Operating License Nos. DPR-39 and DPR-48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47966) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 29, 1996. No significant hazards consideration comments received: No

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

Detroit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan Date of application for amendment: September 5, 1996 (NRC-96-0075), as supplemented by letters dated October 14, October 23, October 29, and October 31, 1996

Brief description of amendment: The amendment revises Technical Specification (TS) 2.1.2 to incorporate cycle-specific safety limit minimum critical power ratios (SLMCPRs) for the core that will be loaded for Cycle 6. In addition, TS 3.4.1.1 is revised to delete the specific SLMCPR number and replace it with a reference to TS 2.1.2.

Date of issuance: November 5, 1996

Effective date: November 5, 1996, with full implementation within 45 days

Amendment No.: 109

Facility Operating License No. NPF-43 Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: September 25, 1996 (61 FR 50342) The letters of October 14, 23, 29, and 31, 1996, provided clarifying information and were not outside the scope of the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 5, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161

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 application for amendments: December 14, 1995, as supplemented by letters dated May 16 and August 29, 1996

Brief description of amendments: The amendments modify the Technical Specifications for diesel generators to incorporate guidance and recommendations contained in NRC Generic Letter (GL) 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' GL 94-01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,'' NUREG-1431, ''Revised Standard Technical Specifications for Westinghouse PWRs,'' and NUREG-1366, ''Improvements to Technical Specifications Surveillance Requirements.''

Date of issuance: October 30, 1996

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

Amendment Nos.: 155 and 147

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

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31175) The August 29, 1996, submittal provided additional information that did not change the scope of the December 14, 1995, 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 October 30, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Duke Power Company, Docket Nos. 50-269, 50-270 and 50-287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South CarolinaDate of application for amendments: August 12, 1996, as supplemented by letter dated September 10, 1996

Brief description of amendments: The amendments revise the Technical Specifications associated with the containment leak-rate tests by implementing 10 CFR Part 50, Appendix J, Option B, for Type A leak-rate testing.

Date of issuance: October 30, 1996

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

Amendment Nos.: 218, 218, 215

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44356) The September 10, 1996, letter provided additional information that did not change the scope of the August 12, 1996, 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 October 30, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: April 15, 1996 (TSCR No. 244)

Brief description of amendment: The amendment revises Specification 5.3.1.B to allow the shield plug and the associated lifting hardware to be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

Date of Issuance: November 7, 1996, to be implemented within 30 days of issuance

Effective date: November 7, 1996

Amendment No.: 187

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20849) The Commission's related evaluation of this amendment and final determination of no significant hazards consideration addressing comments received on the proposed no significant hazards consideration determination are contained in a Safety Evaluation dated November 7, 1996.No significant hazards consideration comments received: Yes.

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

GPU Nuclear, Inc., Docket No. 50-320, Three Mile Island Nuclear Station, Unit No. 2, (TMI-2), Dauphin County, Pennsylvania

Date of application for amendment: February 6, 1995

Brief description of amendment: This amendment revised the Technical Specifications by extending the surveillance interval to demonstrate operability of the containment airlocks from quarterly to annually and to decrease the personnel exposure with implementing the surveillance.

Date of issuance: October 24, 1996

Effective date: October 24, 1996

Amendment No.: 51Possession-Only License No. DPR-73: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28616) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105

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

Date of amendment request: May 1, 1995, as supplemented by letters dated June 22, August 28, November 22, and December 19, 1995, and January 4, 8 (two letters), and 23, June 27, July 9, August 8, and September 23, 1996.

Brief description of amendments: The amendments allowed extension of the standby diesel generator allowed outage time to 14 days, and extension of the essential cooling water loop and the essential chilled water loop allowed outage times to 7 days. The amendments also added to Administrative Controls a description of the Configuration Risk Management Program (CRMP) used to assess changes in core damage probability resulting from applicable plant configurations.

Date of issuance: October 31, 1996

Effective date: October 31, 1996, to be implemented within 30 days

Amendment Nos.: 85 and 72

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

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40019) The additional information contained in the supplemental letters dated August 8 and September 23, 1996, were clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: February 22, 1996, and supplemented July 22, 1996

Brief description of amendments: The amendments revise the administrative controls section of the technical specifications to change the operator license requirements for operations management.

Date of issuance: October 29, 1996

Effective date: October 29, 1996, with full implementation within 45 days

Amendment Nos.: 212 and 197

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

Date of initial notice in Federal Register: March 27, 1996 (61 FR 13527) The July 22, 1996, submittal was more restrictive than the original submittal and did not change the staff's original no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 29, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, ConnecticutDate of application for amendment: August 27, 1996

Brief description of amendment: The Technical Specification (TS) amendment clarifies the limiting condition for operation and surveillance requirements to ensure that the appropriate number of charging pumps and high pressure safety injection pumps are operable for reactivity control and reactor coolant system (RCS) makeup requirements, while also limiting the number of operable pumps to ensure that the low temperature overpressure limits will not be exceeded in the event of a mass addition to the RCS during shutdown conditions. The TS Bases remain unchanged as the result of this amendment.

Date of issuance: October 25, 1996

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

Amendment No.: 205

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

Date of initial notice in Federal Register: September 20, 1996 (61 FR 49498) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 1996No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New YorkDate of application for amendment: March 22, 1996, as supplemented October 11, 1996

Brief description of amendment: The amendment proposed changes to the Technical Specifications to establish operability requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with Boiling Water Reactor Owners Group long-term solution Option I-D. Editorial changes are also made to support the revised specifications, improve readability of Bases sections, and enhance the presentation of requirements for single loop operation.

Date of issuance: October 30, 1996

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

Amendment No.: 236

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20854) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 1996No 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.

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

Date of application for amendment: May 30, 1996, as supplemented by letter dated October 11, 1996

Brief description of amendment: The amendment proposes to eliminate selected response time testing requirements for certain sensors and specified loop instrumentation.

Date of issuance: October 28, 1996

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

Amendment No.: 235

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34896) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 28, 1996.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.

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey Date of application for amendments: July 12, 1996, as supplemented September 12, 1996

Brief description of amendments: The amendments revise Technical Specification Table 3.3-3, ''Engineered Safety Feature Actuation System Instrumentation,'' to clarify the setpoint for the interlock designated P-12.

Date of issuance: November 4, 1996

Effective date: Both units, as of date of issuance, to be implemented within 30 days.

Amendment Nos. 185 and 167

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

Date of initial notice in Federal Register: July 23, 1996 (61 FR 38229) The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the Federal Register notice.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 4, 1996.No significant hazards consideration comments received: No

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

Dated at Rockville, Maryland, this 13th day of November 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga,

Director, Division of Reactor Projects - I/II,Office of Nuclear Reactor Regulation

[FR Doc. 96–29584 Filed 11–18–96; 8:45 am]

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Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a draft of a guide planned for its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide, temporarily identified by its task number, DG–1052 (which should be mentioned in all correspondence concerning this draft guide), is titled ''Time Response Design Criteria for Safety-Related Operator Actions.'' The guide will be in Division 1, ''Power Reactors.'' This regulatory guide is being developed to provide methods acceptable to the NRC staff for developing and applying timing criteria for safety-related operator actions. This guide endorses the American National Standards Institute/American Nuclear Society standard ANSI/ANS–58.8–1994, ''Time Response Design Criteria for Safety-Related Operator Actions.''

This draft guide DG–1052 supersedes DG–1040, which was issued in June 1995 with the same title. DG–1052 has been developed as a result of comments received on DG–1040 and review by the NRC's Advisory Committee on Reactor Safeguards (ACRS). Based on the information presented in DG–1040, the ACRS, in its letter dated November 14, 1995, to the NRC Executive Director for Operations,[1] has raised the following concerns: (1) There is no technical basis for the estimates of minimum times for operator actions in ANSI/ANS–58.8–1994; (2) comparison of the recommended times with results from exercises on plant simulators does not demonstrate that these times are appropriately conservative; (3) endorsement of the standard is not the appropriate way to resolve Generic Safety Issue B–17; and (4) the standard does not address operator response times for advanced nuclear power plants.

The draft guide has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on Draft Regulatory Guide DG–1052 and on the ACRS concerns. Comments may be accompanied by additional relevant information or supporting data. Written comments may be submitted 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. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by January 24, 1997.

Although a time limit is given for comments on this draft guide, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

Comments may be submitted electronically, in either ASCII text or Wordperfect format (version 5.1 or later), by calling the NRC Electronic Bulletin Board on FedWorld. The bulletin board may be accessed using a personal computer, a modem, and one of the commonly available communications software packages, or directly via Internet.

If using a personal computer and modem, the NRC subsystem on FedWorld can be accessed directly by dialing the toll free number: 1–800–

[1] Copies of this letter are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL–6, Washington, DC 20555; telephone (202) 634–3273; fax (202) 634–3343.