present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, the Westinghouse Electric Corporation, their consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the scheduling of sessions which are open to the public, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Mr. Paul A. Boehnert (telephone 301/415–8065) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes in the proposed agenda, etc., that may have occurred.

Dated: December 27, 1995.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 96–00048 Filed 1–2–96; 8:45 am]

BILLING CODE 7590–01–P

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 December 11, 1995, through December 20, 1995. The last biweekly notice was published on December 20, 1995 (60 FR 65672).

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 February 2, 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, 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.

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

Date of amendments request: November 1, 1995, as supplemented on December 1, 1995

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Power Plant, Unit Nos. 2 and 3, Technical Specifications (TSs) and supporting TS Bases relating to the electrical distribution system. The changes are necessary to accommodate the installation of a new safety-related emergency diesel generator (EDG) and a non-safety EDG. The non-safety EDG will be used as an alternate air conditioning source of power in case of a station blackout. In addition to reflecting the new plant configuration, the proposed TSs also reflect the upgraded electrical capacities of the existing EDGs, increased fuel oil storage, and fire protection system for the new EDG building.

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

The Engineered Safety Features (ESF) electrical system provides a reliable source of electrical power to the 4.16 kV ESF busses to operate the necessary accident mitigation equipment, should offsite power be lost. The proposed change to Units 1 and 2 Technical Specifications was prompted by two significant modifications to this system - the addition of No. 1A Emergency Diesel Generator (EDG) and the upgrade of the electrical capacity of two of the three existing Fairbanks Morse EDGs. The addition of No. 1A EDG provides the plant with an ESF electrical system configuration consisting of two EDGs dedicated to each unit, thereby eliminating reliance upon a ''swing'' diesel capable of being aligned to either unit. The four-EDG configuration provides a greater degree of flexibility when an EDG is being overhauled or tested during refueling outages. The increased electrical capacity of the existing Fairbanks Morse EDGs will give the operators greater flexibility in the choice of discretionary loads for the mitigation of accidents. Both modifications necessitate changes to the Technical Specifications.

The ESF electrical system, including the four EDGs, is used to mitigate the consequences of an accident. The design of the new No. 1A EDG is such that incorporation of this EDG into the existing ESF electrical system does not result in this system becoming an accident initiator. Furthermore, the modification to upgrade the capacity of the existing EDGs will enhance the plant operators' ability to mitigate accidents by allowing greater flexibility in the choice of discretionary loads, but will not change the configuration of the ESF electrical system or any support systems such that the EDGs would become an accident initiator. Therefore, the proposed change would not increase the probability of an accident previously evaluated.

The addition of the safety-related No. 1A EDG to the ESF electrical system will enhance the ability to provide reliable electric power during all modes of operation and shutdown conditions of the plant. Number 1A EDG and its support systems are designed such that failure of a single component will not prevent the capability to safely shut down the plant and to maintain the plant in a safe shutdown condition. Furthermore, non-safety-related systems associated with No. 1A EDG are designed so that their failure will not result in the loss of function of any safety-related system. The design of the Fire Protection System in the Diesel Generator Building meets the Codes and Standards specified in the mechanical and instrumentation and controls design reports, previously approved by the

Commission. Inclusion of components from these systems into the Technical Specifications is consistent with Calvert Cliff's current licensing basis. The proposed Technical Specifications will demonstrate the reliability and capability of No. 1A EDG and the upgraded Fairbanks Morse EDGs to perform their accident mitigation function. Implementation of the proposed Technical Specifications will not reduce the ability of the EDGs to perform their safety functions. The increased volume of fuel oil necessary to support operation of No. 1A EDG and the upgraded Fairbanks Morse EDGs will not adversely impact the ability of any systems to perform their safety functions. The auxiliary systems which required modification or analysis to support the upgraded ratings of the Fairbanks Morse EDGs will not adversely impact operation of any other plant systems necessary to mitigate the consequences of an accident. Based on the above, the proposed change would not increase the consequences of an accident previously evaluated.

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

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

The proposed change adds Surveillance Requirements, Limiting Conditions for Operation, and Action Statements to reflect the addition of a new EDG to the ESF electrical system, and upgrades the electrical capacity of the existing Fairbanks Morse EDGs. This change does not add any new equipment, modify any interfaces with any existing equipment, or change the equipment's function, or the method of operating the equipment to be modified. The system will continue to operate in the same manner as before the capacity upgrades were implemented. The additional fuel oil required to support the capacity upgrades will be stored in the existing Seismic Category I fuel oil storage tanks. The modified EDGs will continue to serve a function as accident mitigators, and will not become an initiator of any accident.

The NRC has reviewed the design of the new EDG, its attendant support systems and the new EDG Building, and concurs with Baltimore Gas and Electric Company's determination that the design satisfies the design requirements for a safety-related EDG. Number 1A EDG is a tandem engine-single generator set, and is physically very different from the existing single engine-generator Fairbanks Morse EDGs. However, the 4.16 kV three-phase rated electrical output is the same as that provided by the Fairbanks Morse EDGs to the other ESF busses. The excess capacity of No. 1A EDG will allow the operators greater flexibility in choosing post-accident discretionary loads, but will not cause any detrimental effects to the ESF busses or the equipment served by those busses. Operation of No. 1A EDG in accordance with these proposed Technical Specifications will not jeopardize the operation of any other plant systems. The design of the Fire Protection System in the Diesel Generator Building meets the Codes and Standards specified in the mechanical, and instrumentation and controls design reports, previously approved by the Commission. Inclusion of components from these systems into the Technical Specifications is consistent with Calvert Cliffs current licensing basis. Furthermore, locating No. 1A EDG and its fuel oil supply in a separate Category I building provides additional assurance that this equipment will not become an initiator of any accident.

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

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

The safety function of the EDGs and the ESF electrical system is to provide a reliable source of electrical power to the safety-related busses to operate the necessary accident mitigation equipment, should offsite power be lost. The margin of safety associated with this safety function is two-fold: (1) a level of redundancy must be designed into the EDGs and the ESF electrical system such that the single failure criteria is met; and (2) the power supplied to the ESF electrical system by the EDGs must be sufficient to power the necessary accident mitigation equipment, should offsite power be lost.

The addition of No. 1A EDG provides the plant with an ESF electrical system configuration consisting of two EDGs dedicated to each unit, thereby eliminating reliance upon a swing diesel capable of being aligned to either unit. In the current configuration, the facility meets the single failure criteria on a ''per site'' basis. However, as a result of the new four-EDG configuration, each unit will have redundant diesel generators to supply power to redundant safety-related equipment required for safe shutdown or accident mitigation. The revised Fuel Oil System configuration and the minimum fuel oil volume to be maintained in the fuel oil tanks supports the safety function of the EDGs, while maintaining the margin of safety associated with this equipment. Altogether, the new four-EDG configuration may be considered an increase in the margin of safety.

Inclusion of Surveillances for the Fire Protection System components into the Technical Specifications is consistent with Calvert Cliffs current licensing basis, and ensures that adequate fire detection and suppression capability is available to identify and extinguish fires in the Diesel Generator Building, thereby reducing the potential for damage to No. 1A EDG and its auxiliaries. The Diesel Generator Building and its Fire Protection System is designed so that smoke and heat from a fire in that building will not impact the redundant safety-related Emergency Diesel Generator in the Auxiliary Building.

At the completion of the modifications to increase the capacities of the Unit 2 EDGs and to install the new No. 1A EDG, we will have diesel generators with more available margin than currently exists. This will provide the operators with more flexibility during conditions where the diesel generators are providing onsite power. The higher electrical capacities will result in an increase in the margin between the EDGs' electrical capacities and the electrical power required to operate safety-related equipment required for safe shutdown or accident mitigation. Therefore, these modifications may be considered an increase in the margin of safety.

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

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

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

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

Date of amendments request: November 30, 1995

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Technical Specifications (TSs) to allow the installation of tube sleeves as an alternative to plugging for repairing steam generator (SG) tubes. The proposed changes to TS 3/4.4.5, ''Steam Generators,'' and their supporting Bases would permit tube sleeving repair techniques developed by Westinghouse Electric Corporation and ABB Combustion Engineering, Inc., to be used as a repair method for the SGs at the Calvert Cliffs site.

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

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

The implementation of the proposed steam generator tube sleeving has been reviewed for impact on the current CCNPP [Calvert Cliffs Nuclear Power Plant] licensing basis.

Since the sleeve dimensions, materials, and connecting joints to the existing tube are designed to the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, the proposed sleeving repair acts as an in-kind substitution for the original steam generator tubing. The applicable design criteria for the sleeves conform to the stress limits and margins of safety of Section III of the ASME Code. Safety factors of 3 for normal operation and 1.5 for accident conditions were applied to the design. Mechanical testing using the ASME Code stress allowables has been performed in support of the design. Based on the results of Westinghouse and ABB-Combustion Engineering analytical and test programs, the sleeves fulfill their intended function as leak tight structural members and meet or exceed all design criteria.

Evaluation of the proposed sleeved tubes indicates no detrimental effects on the sleeve or sleeve-tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions or transients as may be experienced at CCNPP. Corrosion testing of sleeve-tube assemblies indicate no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The installation of the proposed sleeves is controlled via the sleeving vendors' proprietary processes and equipment. The ABB Combustion Engineering process has been in use since 1984, and has been implemented 24 times for the installation of over 4,200 sleeves. The Westinghouse process has been in use since 1988, and approximately 12,000 laser welded sleeves have been installed between 1988 and 1994. The CCNPP steam generator design was reviewed and found to be compatible with both installation processes and equipment.

The implementation of the proposed sleeves has no significant effect on either the configuration of the plant, or the manner in which it is operated. The hypothetical consequences of failure of the sleeved tube is bounded by the current steam generator tube rupture analysis described in Section 14.15 of the Calvert Cliffs Updated Final Safety Analysis Report.

Therefore, BGE [Baltimore Gas and Electric] has concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) [The proposed amendment] would not create the possibility of a new or different kind of accident from any other accident previously evaluated.

As discussed above, the structural integrity, thermal characteristics, and material properties of the proposed sleeves are consistent with the existing plant steam generators. Therefore, the functions of the steam generators will not be significantly affected by the installation of the proposed sleeves. In addition, the proposed sleeves do not interact with any other plant systems. The continued integrity of the installed sleeve is periodically verified by the Technical Specification requirements. The implementation of the proposed sleeves has no significant effect on either the configuration of the plant, or the manner in which it is operated.

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

(3) [The proposed amendment] would not involve a significant reduction in a margin of safety.

The repair of degraded steam generator tubes via the use of the proposed sleeves has been confirmed to restore the structural integrity of the faulted tube under normal operating and postulated accident conditions. The design safety factors utilized for the sleeves are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original steam generator design. The repair limit for the proposed sleeves is consistent with that established for the steam generator tubes. The design of the sleeve to tube joints is verified by testing to preclude significant leakage during normal and postulated accident conditions. Use of the previously identified design criteria and design verification testing assures that the margin to safety with respect to the implementation of the proposed sleeves is not significantly different from the original steam generator tubes.

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

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

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

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

Date of amendments request: December 7, 1995

Description of amendments request: The proposed amendments would change the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Technical Specifications (TSs) by adding an analysis technique to the list of approved core operating limits analytical methods. Specifically, these amendments would add the convolution analysis technique to the list of approved methodologies in TSs 6.9.1.9.b. The convolution analysis technique has already been reviewed and approved by the NRC staff and the supporting safety evaluation was provided to the licensee by an NRC letter dated May 11, 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:

The change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration in that operation of the facility in accordance with the proposed amendment:

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

The proposed change is to add the convolution analysis technique previously approved by the NRC to the list of approved methodologies in Calvert Cliffs' Unit 1 and 2 Technical Specifications. By letter dated November 1, 1994, Baltimore Gas and Electric Company (BGE) requested approval to use the ABB/Combustion Engineering (ABB/CE) convolution technique for determining the values in the Calvert Cliffs Core Operating Limits Report (COLR) related to the pre-trip main steam line break event. Approval was given by the NRC in their letter dated May 11, 1995. The addition of this technique to the list of approved analytical methods in Technical Specification 6.9.1.9.b is simply intended to identify it as an approved methodology. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is to add the convolution analysis technique previously approved by the NRC to the list of approved methodologies in Calvert Cliffs' Unit 1 and 2 Technical Specifications. By letter dated November 1, 1994, BGE requested approval to use the ABB/CE convolution technique for determining the values in the Calvert Cliffs COLR related to the pre-trip main steam line break event. Approval was given by the NRC in their letter dated May 11, 1995. The addition of this technique to the list of approved analytical methods in Technical Specifications 6.9.1.9.b is simply intended to identify it as an approved methodology. Therefore, the change would not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in the margin of safety.

The proposed change is to add the convolution analysis technique previously approved by the NRC to the list of approved methodologies in Calvert Cliffs' Unit 1 and 2 Technical Specifications. By letter dated November 1, 1994, BGE requested approval to use the ABB/CE convolution technique for determining the values in the Calvert Cliffs COLR related to the pre-trip main steam line break event. Approval was given by the NRC in their letter dated May 11, 1995. The addition of this technique to the list of approved analytical methods in Technical Specification 6.9.1.9.b is simply intended to identify it as an approved methodology. Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

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

Date of amendment request: December 7, 1995

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 3.4.5 and 3.4.6.2 and their Bases to maintain voltage-based steam generator tube repair criteria for the tube support plate elevations beyond the current cycle of operation. The proposed amendment would implement a 2.0 volt repair limit to replace a 1.0 volt repair limit which was approved on an interim basis for only the current fuel cycle by License Amendment No. 184 [issued February 3, 1995]. The proposed amendment would also include changes in addition to those incorporated by License Amendment No. 184 to reflect the guidance provided in NRC Generic Letter (GL) 95-05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.''

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?

Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate (TSP). Test data indicates that tube burst cannot occur within the TSP, even for tubes which have 100% throughwall electric discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Since tube-to-TSP proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition, main steamline break (MSLB) pressure differential. As previously stated, the Regulatory Guide (RG) 1.121 criterion requiring maintenance of a safety factor of 1.43 times the MSLB pressure differential on tube burst is satisfied by 7/8'' diameter tubing with bobbin coil indications with signal amplitudes less than 8.82 volts, regardless of the indicated depth measurement.

The upper voltage repair limit (V_{url}) will be determined prior to each outage using the most recently approved NRC database to determine the tube structural limit (V_{sl}). The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{nde}) and growth (V_{gr}) to establish V_{url}. Using Generic Letter (GL) 95-05 and growth allowances for an example, the NDE uncertainty component of 20% and a voltage growth allowance of 30% per full power year can be utilized to establish a V_{url} of 5.9 volts. The 20% NDE uncertainty represents a square-root-sum-of-the-squares (SRSS) combination of probe wear uncertainty and analyst variability. The degradation growth allowance should be an average growth rate or 30% per effective full power year, whichever is larger. This growth allowance is conservative for BVPS-1 [Beaver Valley Power Station, Unit No. 1] as the percent voltage growth rates have decreased for each of the last three inspections.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated MSLB outside of containment but upstream of the main steam isolation valve (MSIV) represents the most limiting radiological condition relative to the plugging criteria. In support of implementation of the revised plugging limit, analyses will be performed to determine whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary-to-secondary leakage would result in postulated site boundary and control room doses exceeding 10 CFR 100, and 10 CFR 50, Appendix A, GDC-19 requirements, respectively. A separate calculation has determined the maximum allowable MSLB leakage limit in a faulted loop. This limit was calculated using the technical specification reactor coolant system (RCS) Iodine-131 activity level of 1.0 microcuries per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800. The projected MSLB leakage rate calculation methodology prescribed in Section 2.b of GL 95-05 will be used to calculate the end-of-cycle (EOC) leakage. Projected EOC voltage distribution will be developed using the most recent EOC eddy current results and considering an appropriate voltage measurement uncertainty. The log-logistic probability of leakage correlation will be used to establish the MSLB leakrate used for comparison with the faulted loop allowable limit. Due to the relatively low voltage levels of indications at BVPS-1 and low voltage growth rates, it is expected that the calculated leakage values will not exceed this limit. Therefore, as implementation of the 2.0 volt voltage-based plugging criteria at BVPS-1 does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report].

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

Implementation of the proposed steam generator tube 2.0 volt plugging limit does not introduce any significant changes to the plant design basis. Use of the 2.0 volt plugging limit does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations as no outside diameter stress corrosion cracking (ODSCC) is occurring outside the thickness of the tube support plates. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging limit has been applied (during all plant conditions).

Duquesne Light Company will continue to implement a maximum primary-to-secondary leakage rate limit of 150 gpd [gallons per day] per steam generator to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit provides for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. RG 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded.

The single through-wall crack lengths that result in tube burst at 1.43 times the MSLB pressure differential and the MSLB pressure differential alone are approximately 0.57 inch and 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of 0.41 inch long cracks at nominal leak rates and 0.62 inch long cracks at the lower 95% confidence level leak rates. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during MSLB conditions, the leakage from the maximum permissible crack must preclude tube burst at MSLB conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for MSLB conditions using the lower 95% leakrate data. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection. Analyses have shown that only a small percentage of the TSPs are deflected greater than the TSP thickness during a postulated MSLB.

As steam generator tube integrity upon implementation of the 2.0 volt plugging limit continues to be maintained through inservice inspection and primary-to-secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The use of the voltage-based bobbin probe tube support plate elevation plugging criteria at BVPS-1 maintains steam generator tube integrity commensurate with the criteria of RG 1.121. This guide describes a method acceptable to the Commission for meeting GDCs [General Design Criterion] 14, 15, 30, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the proposed criteria, even under the worst case conditions, the occurrence of ODSCC [Outside Diameter Stress Corrosion Cracking] at the tube support plate elevations is not expected to lead to a steam generator tuberupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the tube support plate elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of loss-of-coolant-accident (LOCA) + safe shutdown earthquake (SEE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse. There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to complete through-wall cracks during tube deformation or collapse.

The results of an analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation. Since the leak-before-break methodology is applicable to BVPS-1 reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. Analysis results provided in WCAP-14122, dated July 1994, demonstrate that no tubes were subject to deformation or collapse. No tubes have been excluded from application of the subject voltage-based steam generator plugging criteria.

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage-based tube plugging criteria of 2.0 volts is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the tube support plate elevations, and rotating pancake coil inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

As noted previously, implementation of the tube support plate intersection voltage-based plugging criteria will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, the implementation of the 2.0 volt plugging limit will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the UFSAR or any BASES of the plant technical specifications.

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

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: December 15, 1995

Description of amendment request: The proposed amendments would (1) revise Technical Specifications (TSs) 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 3/4.6.1.6, and associated Bases, (2) delete TS 6.9.2.g, and (3) add a new TS 6.17. The proposed changes would make the TSs consistent with Option B of recently revised Appendix J of 10 CFR Part 50 and the implementing guidance of Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' dated September 1995. Option B of Appendix J permits licensees to implement a performance based option rather than the previous prescriptive requirements now contained in Appendix J as Option A. The proposed amendments would remove from the TSs the prescriptive requirements of Option A concerning test frequencies and test methodology and would also include minor administrative and editorial changes to add consistency between the Bases and the TSs and to provide additional clarification.

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?

Containment leakage is not an accident initiator. The proposed amendment does not add or modify any existing plant equipment. Therefore there is no increase in the probability of an accident previously evaluated.

The consequences of an accident previously evaluated are not significantly increased. The proposed changes do not affect the assumptions, parameters or result of any Updated Final Safety Analysis (UFSAR) accident analyses. The containment leakage rate will continue to be maintained within the limit assumed in the accident analysis for a Design Basis Accident (DBA). The proposed changes do not modify the response of the containment during a DBA. The proposed amendment will continue to ensure that the ability of the containment structure, including the containment air locks, to limit leakage from a DBA is demonstrated using test methodologies and guidance on test frequencies that have been determined to be acceptable to meet the requirements of 10 CFR 50, Appendix J, Option B.

The potential increase to overall accident risk due to the containment leak tightness decreasing between extended testing intervals and the resulting potential increased radioactivity release to the environment during a DBA has been determined to be minimal based on the findings of NUREG 1493 titled ''Performance-Based Containment Leak-Test Program.'' In addition, due to the performance based nature of 10 CFR 50 Appendix J, Option B, the extended test intervals are utilized only when the component(s) have demonstrated an acceptable performance history. Therefore, a significant decrease in containment leak tightness between extended test intervals is not expected as a result of this proposed change.

Based on the above discussion, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical changes to the plant or changes in plant operating configuration. The proposed amendment involves changes to plant programs and administrative requirements used in determining acceptable containment performance. The performance of plant systems, including the containment structure, during plant operation remains unchanged.

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

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

The margin of safety is not significantly reduced by this proposed change. The acceptance criteria for ''as left'' measured containment leakage rates is not being increased as result of this proposed amendment. For Beaver Valley Power Station (BVPS) Unit No. 1 only, the ''as found'' maximum allowable overall Type A leakage rate is being slightly increased. However, the slight increase does not exceed the value assumed in accident analysis for containment leakage during a DBA due to changing the acceptance criteria from less than to less than or equal to. The margin between the acceptable ''as left'' measured overall Type A containment leakage rate and the leakage rate assumed in the accident analysis is not being decreased.

The maximum ''as found'' allowable overall Type A leakage rate remains unchanged for BVPS Unit No. 2. The margin between the acceptable ''as left'' measured overall Type A containment leakage rate and the leakage rate assumed in the accident analysis is also not being decreased.

The maximum allowable measured combined Type B and C leakage rate is not being increased above the current limits.

The maximum peak containment pressure following a DBA remains unchanged. The containment depressurization time following a DBA remains unchanged. The calculated offsite dose consequences of a DBA remains unchanged.

The proposed amendment continues to ensure reactor containment system reliability by periodic testing in compliance with 10 CFR 50, Appendix J, Option B. The extension of Type A, B and C test frequencies permitted by 10 CFR 50 Appendix J, Option B, is not expected to result in a significant decrease in containment leak tightness between test intervals. Due to the performance based nature of 10 CFR 50 Appendix J, Option B, the extended test intervals are utilized only when the component(s) have demonstrated an acceptable performance history. Therefore, a significant decrease in containment leak tightness between extended test intervals is not expected as a result of this proposed change.

The changes which are either administrative or editorial in nature will not reduce the margin of safety because they have no impact on any safety analysis assumptions.

Therefore, based on the above discussion, it can be concluded that the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 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: May 19, 1995, as supplemented by letter dated December 7, 1995.

Description of amendment request: May 19, 1995, submittal requested to modify Action Statement for Technical Specification (TS) 3.6.4.2 for the hydrogen recombiners. It also requested to make the surveillance requirements for hydrogen recombiners consistent with NUREG-1432, ''Standard Technical Specifications Combustion Engineering Plants.'' The December 7, 1995, letter withdrew the request to change the Action Statement for TS 3.6.4.2.

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 response is predicated on the following technical bases: (1) the current licensing basis of record establishes that only one recombiner system is required to maintain hydrogen concentration below 4%, (2) the proposed technical specification changes are conservative when compared with the recommendations of Regulatory Guide 1.7, (3) short term post LOCA hydrogen generation is less than 1%, (4) long term post LOCA hydrogen generation is less than the flame propagation limit, which according to Regulatory Guide 1.7 would not result in adverse effects to containment systems, and (5) a design basis LOCA without long term hydrogen control would produce pressures below the containment design pressure.... Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change will not alter the configuration or operation of any other plant system or component. The change does not involve any change to the operational design or limits of any other plant systems or components. Thus, no new failure modes are introduced or associated with the proposed change. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will have no adverse impact on the protective boundaries, safety limits, or margin or safety. There are no limits or margins of safety being revised for any systems, components, or protective boundaries.

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

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

Date of amendment request: November 7, 1995

Description of amendment request: Amendment to Technical Specification (TS) 3/4.8.1 ''Electrical Power Systems - AC Sources'' and the associated TS BASES. The proposed amendment would implement selected changes from NUREG 1432, ''Standard Technical Specifications Combustion Engineering Plants,'' Generic Letter (GL) 94-01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,'' and GL 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' The intent of these changes is to increase Emergency Diesel Generator (EDG) reliability by reducing the stresses on the EDGs caused by unnecessary testing. This proposed TS amendment will also relocate the Surveillance Requirements for maintaining the properties of the fuel oil to TS Section 6, ''Administrative Controls.'' These requirements will be implemented as part of the Fuel Oil Testing Program. In addition, the requirement for cleaning the diesel fuel oil storage tanks with a sodium hypochlorite solution or equivalent will be changed to also allow an appropriate mechanical method (such as pressure washing or manual wiping) to be utilized.

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 Standby Diesel Generators do not initiate any accidents, therefore the proposed changes do not increase the probability of an accident previously evaluated. The proposed changes to TS 3/4.8.1 and the associated BASES affect the required actions in response to inoperable offsite and onsite AC sources, Surveillance Requirements for the EDG, and reporting requirements for EDG failures. The majority of the proposed changes are based on the recommendations of NUREG 1432, GL 94-01, and GL 93-05. These proposed changes have been extensively reviewed by the NRC during the preparation of these documents and by Waterford 3 SES during the development of this request for TS amendment. The proposed changes are expected to result in improvements in EDG performance and reduce EDG aging due to excessive testing. The proposed changes will permit the elimination of the unnecessary mechanical stress and wear on the EDGs while ensuring that the EDGs will perform their design function. The elimination of mechanical stress and wear will improve reliability and availability of the EDGs which will have a positive effect on the ability of the EDGs to perform their design function. The proposed changes do not affect the availability or the testing requirements of the offsite circuits.

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

The proposed changes to TS 3/4.8.1 and the associated Bases do not introduce any new modes of plant operation or new accident precursors, involve any physical alterations to plant configurations, or make any changes to system setpoints which could initiate a new or different kind of accident. The proposed changes do not affect the design or performance characteristics of any EDG or its ability to perform its design function. No new failure modes have been defined and no new system interactions have been introduced for any plant system or component. In addition, there have not been any new limiting failures identified as a result of the proposed changes. The proposed changes will eliminate unnecessary EDG testing and will increase EDG reliability and availability. This will have an overall positive affect on plant safety. Accidents concerning loss of offsite power and a single failure (e.g., loss of an EDG) have previously been evaluated. These changes are intended to improve plant safety, decrease equipment degradation, and remove an unnecessary burden on personnel resources by reducing the amount of testing that the TS requires during power operation.

Relocating the diesel fuel oil testing requirements to the Waterford 3 Fuel Oil Testing Program outside of the Technical Specifications is an administrative change only and consequently has no effect on accident probability, consequences, or margin. Also, the proposed cleaning method for the diesel fuel oil storage tanks meets the intent of Regulatory Guide 1.137 and will not result in the degradation of the fuel oil.

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

Under the proposed changes to TS 3/4.8.1 and the associated Bases, the EDGs will remain capable of performing their safety function. The changes do not affect the design or performance of the EDGs, but will increase EDG reliability and availability by reducing the stresses and the effects of aging on the EDG by eliminating unnecessary testing. This will result in an overall increase in plant safety. The ability of the EDGs to perform their safety function will not be degraded. Relocating the diesel fuel oil testing requirements to the Waterford 3 Fuel Oil Testing Program outside of the Technical Specifications is an administrative change only and consequently has no effect on accident probability, consequences, or margin. Also, the proposed cleaning method for the diesel fuel oil storage tanks meets the intent of Regulatory Guide 1.137 and will not result in a reduction in the margin of safety.

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, Louisiana 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

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

Date of amendment request: (TS 93-09) December 8, 1995

Description of amendment request: The proposed change would revise the setpoints and time delays for the auxiliary feedwater loss-of-power and 6.9-kv shutdown board loss-of-voltage and degraded-voltage instrumentation setpoints in Items 6 and 7 of Technical Specification Table 3.3-4, respectively.

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:

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-feedwater 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-T106-0008, 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 thisreview, 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 llH, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: (TS 95-20) December 8, 1995

Description of amendment request: The proposed change would revise Surveillance Requirements 4.6.2.1.1.d and 4.6.2.1.2.b to extend the containment spray nozzle air or smoke flow tests from the present 5-year interval to a 10-year interval, in accordance with Generic Letter 93-05.

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:

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 TS change is consistent with the guidance provided in Generic Letter 93-05. Containment spray (CS) systems' header piping is stainless steel; therefore, corrosion will be negligible during the extended surveillance interval. Since the CS systems' headers are maintained dry, there is no mechanism that could cause blockage of the spray nozzles. Therefore, the nozzles in the CS systems will remain operable, during the 10-year surveillance interval, to mitigate the consequence of an accident previously evaluated. Additionally, clogging or blockage has not been observed during the 5-year surveillance tests that have been performed in the past at SQN. Testing the CS systems' nozzles at the proposed reduced frequency will not increase the probability of occurrence of a postulated accident or the consequences of an accident previously evaluated.

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

The proposed reduced frequency testing of the CS systems' nozzles does not change the manner in which these systems are operated. The reduced testing frequency of the spray nozzles does not generate any new accident precursors. Therefore, the possibility of a new or different kind of accident previously evaluated is not created by the proposed changes in surveillance frequency of the CS system's nozzles.

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

Reduced testing of the CS systems' nozzles does not change the way the systems are operated or the systems' operability requirements. In this application, any additional corrosion of stainless steel piping will be negligible during the extended surveillance interval. Since the CS systems are maintained dry, there is no additional mechanism that could cause blockage of the nozzles. Therefore, the proposed reduced testing frequency is adequate to ensure spray nozzle operability. The surveillance requirements do not affect the margin of safety since the operability requirements of both the CS systems remains unchanged. The existing safety analysis remains bounding. Therefore, there is no reduction in the margin of safety.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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

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

Date of amendment request: December 8, 1995 (TS 95-24)

Description of amendment request: The proposed change would modify various Technical Specification requirements in order to implement the recent rule change to 10 CFR Part 50, Appendix J. The new Appendix J rule (Option B) provides a voluntary performance based testing option for containment leakage rate testing (CLRT). Option B CLRT requirements are based on system and component performance in lieu of compliance with the current prescriptive requirements. Option B allows extension of the integrated leakage rate test (Type A test) frequency based on an acceptable past history. For Type B and Type C local leak rate test, Option B allows extension of the test frequency based on plant-specific experience history of each component and establishes controls to ensure continued performance during extended testing intervals.

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:

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 amendment to SQN TSs is in accordance with Option B to 10 CFR 50, Appendix J. The proposed amendment adds a voluntary performance based option for containment leak rate testing. The changes being proposed do not affect the precursor for any accident or transient analyzed in Chapter 15 of SQN Updated Final Safety Analysis Report. The proposed change does not increase the total allowable primary containment leakage rate. The proposed change does not reflect a revision to the physical design and/or operation of the plant. Therefore, operation of the facility, in accordance with the proposed change, does not significantly affect the probability or consequences of an accident previously evaluated.

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

The proposed amendment to SQN TSs is in accordance with the new performance-based option (Option B) to 10 CFR 50, Appendix J. The changes being proposed will not change the physical plant or the modes of operation defined in the facility license. The proposed changes do not increase the total allowable primary containment leakage rate. The changes do not involve the addition or modification of equipment, nor do they alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to SQN TSs is in accordance with the new option to 10 CFR 50, Appendix J. The proposed option is formulated to adopt performance-based approaches. This option removes the current prescriptive details from the TS. The proposed changes do not affect plant safety analyses or change the physical design or operation of the plant. The proposed change does not increase the total allowable primary containment leakage rate. Therefore, operation of the facility, in accordance with the proposed change, does not involve a significant reduction in the margin of safety.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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

Tennessee Valley Authority, Docket No. 50-328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of amendment request: December 12, 1995 (TS 95-23)

Description of amendment request: The proposed change would incorporate new requirements associated with steam generator tube inspections and repair. The new requirements would establish alternate steam generator tube plugging criteria at the tube support plate intersections.

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:

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.

Testing of model boiler specimens for free-span tubing (no tube support plate restraint) at room temperature conditions shows burst pressures in excess of 5,000 pounds per square inch (psi) for indications of outer diameter stress corrosion cracking with voltage measurements as high as 19 volts. Burst testing performed on intersections pulled from SQN with up to a 1.9-volt indication shows measured burst pressure in excess of 6,600 psi at room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5-volt indications shows burst pressures in excess of 5,200 psi at room temperatures. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the safety-factor requirements of NRC Regulatory Guide (RG) 1.121.

Tube burst criteria are inherently satisfied during normal operating conditions because of the proximity of the tube support plate (TSP). Since tube-to-tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics that maintain a margin of safety of 1.43 times the bounding faulted condition steam line break (SLB) pressure differential. During a postulated SLB, the TSP has the potential to deflect during blowdown following a main SLB, thereby uncovering the TSP intersections.

Based on the existing database, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the SLB pressure differential on tube burst is satisfied by 7/8-inch-diameter tubing with bobbin coil indications with signal amplitudes less than 8.82 volts (WCAP-13990), regardless of the indicated depth measurement. A 2.0-volt plugging criterion (resulting in a projected end-of-cycle [EOC] voltage) compares favorably with the 8.82-volt structural limit considering the extremely slow apparent voltage growth rates and few numbers of indications at SQN. Using the established methodology of RG 1.121, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning of cycle (BOC) repair limit that would preclude indications at EOC conditions that exceed the structural limit. The nondestructive examination (NDE) uncertainty component is 20.5 percent, and is based on the Electric Power Research Institute (EPRI) alternate repair criteria (ARC).

Test data indicates that tube burst cannot occur within the TSP, even for tubes that have 100 percent throughwall electro-discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Because of the few number of indications at SQN, the EPRI methodology of applying a growth component of 35 percent per effective full power year (EFPY) will be used. Near-term operating cycles at SQN are expected to be bounded by 1.23 years, therefore, a 43 percent growth component is appropriate. When these allowances are added to the BOC alternate plugging criteria (APC) of 2.0 volts in a deterministic bounding EOC voltage of approximately 3.26 volts for Cycle 7, operation can be established. A 5.56-volt deterministic safety margin exists (8.82 structural limit - 3.26-volt EOC equal 5.56-volt margin).

For the voltage/burst correlation, the EOC structural limit is supported by a voltage of 8.82 volts. Using this structural limit of 8.82 volts, a BOC maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit the existence of EOC indications that exceed the 8.82-volt structural limit. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 8.82 volts can be represented by the expressions:

RL + (0.205 x RL) + (0.43 x RL) = 8.82 volts, or,

the maximum allowable BOC repair limit can be expressed as,

RL = 8.82-volt structural limit/1.64 = 5.4 volts.

This RL (5.4 volts) is the appropriate limit for APC implementation to repair bobbin indications greater than 2.0 volts independent of rotating pancake coil (RPC) confirmation of the indication. This 5.4-volt upper limit for non-confirmed RPC calls is consistent with other recently approved APC programs (Farley Nuclear Plant, Unit 2).

The conservatism of the growth allowance used to develop the repair limit is shown by the most recent SQN eddy current data. Only seven tubes in Unit 2 required repair because of outside diameter stress corrosion cracking (ODSCC) at the TSP intersections.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main

SLB outside of containment, but upstream of the main steam isolation valve (MSIV), represents the most limiting radiological condition relative to the APC. Implementation of the APC will determine whether the distribution of cracking indications at the TSP intersections is projected to be such that primary-to-secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines. A separate analysis has determined this allowable SLB leakage limit to be 3.7 gallons per minute (gpm) in the faulted loop. This limit uses the TS reactor coolant system (RCS) Iodine-131 activity level of 1.0 microcuries per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800. The analysis method is WCAP-14277, which is consistent with the guidance of the NRC generic letter (GL) [95-05] and will be used to calculate EOC leakage. Because of the relatively low number of indications at SQN, it is expected that the actual leakage values will be far less than this limit. Additionally, the current Iodine-131 levels at SQN range from about 25 to 100 times less than the TS limit.

Application of the criteria requires the projection of postulated SLB leakage, based on the projected EOC voltage distribution for Cycle 8 operation. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Data indicates that a threshold voltage of 2.8 volts would result in throughwall cracks long enough to leak at SLB condition. The GL requires that all indications to which the APC are applied must be included in the leakage projection. Tube pull results from another plant with 7/8-inch tubing with a substantial voltage growth database have shown that tube wall degradation of greater than 40 percent throughwall was readily detectable either by the bobbin or RPC probe. The tube with maximum throughwall penetration of 56 percent (42 average) had a voltage of 2.02 volts. The SQN Unit 1 pulled tube had a 1.93-volt indication with a maximum depth of 91 percent and did not leak at SLB condition. Based on the SQN pulled tube and industry pulled tube data supporting a lower threshold for SLB leakage of 2.8 volts, inclusion of all APC intersections in the leakage model is quite conservative. The ODSCC occurring at SQN is in its earliest stages of development. The conservative bounding growth estimations to be applied to the expected small number of indications for the upcoming inspection should result in very small levels of predicted SLB leakage. Historically, SQN has not identified ODSCC as a contributor to operational leakage.

In order to assess the sensitivity of an indication's BOC voltage to EOC leakage potential, a Monte Carlo simulation was performed for a 2.0-volt BOC indication.

The maximum EOC voltage (at 99.8 percent cumulative probability) was found to be 4.8 volts. The leakage component from an indication of this magnitude, using the EPRI leakage model, is 0.028 gpm.

Therefore, as implementation of the 2.0-volt APC does not adversely affect steam generator (S/G) tube integrity and implementation will be shown to result in acceptable dose consequences, the proposed amendment does not result in 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 previously analyzed.

Implementation of the proposed S/G tube APC does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism that could result in an accident outside of the region of the TSP elevations; no ODSCC is occurring outside the thickness of the TSP. Neither a single or multiple tube rupture event would be expected in a S/G in which the plugging criteria is applied (during all plant conditions).

TVA will implement a maximum leakage rate limit of 150 gallon per day per S/G to help preclude the potential for excessive leakage during all plant conditions. The SQN TS limits on primary-to-secondary leakage at operating conditions include a maximum of 0.42 gpm (600 gallons per day [gpd]) for all S/Gs, or, a maximum of 150 gpd for any one S/G. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown is based upon leak-before-break considerations to detect a free-span crack before potential tube rupture during faulted plant conditions. The 150-gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. RG 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at faulted conditions maximum pressure differential. A voltage amplitude of 8.82 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95 percent prediction limit on the burst correlation coupled with 95/95 lower tolerance limit material properties. Alternate crack morphologies can correspond to 8.82 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the ''longest permissible crack'' for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at 1.43 times the SLB pressure differential and the SLB pressure differential alone are approximately 0.57 inch and 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of 0.4-inch-long cracks at nominal leak rates and 0.6-inch-long cracks at the lower 95 percent confidence level leak rates. Since tube burst is precluded during normal operation because of the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during SLB conditions, the leakage from the maximum permissible crack must preclude tube burst at SLB conditions. Thus, the 150-gpd limit provides for plant shutdown before reaching critical crack lengths for SL-conditions. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncover will provide benefit to the burst capacity of the intersection.

As S/G tube integrity upon implementation of the 2.0-volt APC continues to be maintained through in-service inspection and primary-to-secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The use of the voltage based APC at SQN is demonstrated to maintain S/G tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC Staff for meeting General Design Criteria (GDC) 14, 15, 31, and 32 by reducing the probability or the consequences of S/G tube rupture. This is accomplished by determining the limiting conditions of degradation of S/G tubing, as established by in-service inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst-case conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a S/G tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and radiological consequences are not adversely impacted.

In addressing the combined effects of loss-of-coolant accident (LOCA), plus safe shutdown earthquake (SSE) on the S/G component (as required by GDC 2), it has been determined that tube collapse may occur in the S/Gs at some plants. This is the case as the TSP may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate because of the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with S/G tube collapse. First, the collapse of S/G tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA, which in turn, may potentially increase peak clad temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the SQN reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. LOCA loads for the primary pipe breaks were used to bound the conditions at SQN for smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in S/G tube collapse or significant deformation. The LOCA, plus SSE tube collapse evaluation performed for another plant with Series 51 S/Gs using bounding input conditions (large-break loadings), is applicable to SQN. Therefore, at SQN, no tubes will be excluded from using the voltage repair criteria due to deformation of collapse of S/G tubes following a LOCA plus an SSE. Additional supporting information relative to NRC review of J.M. Farley Nuclear Plant was provided in Enclosure 5, Item 3 of TVA's submittal dated September 7, 1995 (TAC No. M92961).

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria of 2.0 volt is supplemented by: (1) enhanced eddy current inspection guidelines to provide consistency in voltage normalization, (2) a 100 percent eddy current inspection sample size at the TSP elevations, and (3) RPC inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

As noted previously, implementation of the TSP elevation plugging criteria will decrease the number of tubes that must be repaired. The installation of S/G tube plugs reduces the RCS flow margin. Thus, implementation of the alternate plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin of safety.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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

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

Date of amendment request: November 21, 1995

Brief description of amendments: The proposed amendments would modify the Comanche Peak Steam Electric Station (CPSES) Units 1 and 2 Technical Specifications (TS) to allow the containment personnel airlock (PAL) doors to remain open during movement of irradiated fuel and during core alterations.

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

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

The proposed change allows the PAL doors for containment to remain open during the movement of irradiated fuel and core alterations. Whether or not the PAL doors are open does not effect the movement of fuel, the strict compliance with the procedures governing refueling operations, or the integrity of fuel assemblies. The position of the airlock doors cannot, in itself, be the initiating event in any accident. The probability of a fuel handling accident is not changed.

The consequences of leaving the airlock doors open during this accident are bounded by the existing analysis, provided the fuel handling accident assumptions are maintained (e.g. 100 hours after reactor shutdown and the water level remains 23 feet above the fuel). The existing analysis postulates the limiting fuel handling accident to occur in the Fuel Building with no credit taken for barrier or filtration. This accident analysis envelopes the proposed change for a fuel handling accident occurring in the Containment Building.

Were a fuel handling accident to occur with the PAL doors open, the impact would be minimal. Pressure is expected to be essentially equalized across the door with little air flow either into or out of containment. Based on transport time from the location of the accident to the PAL, little, if any, radioactive material is expected to escape containment via the PAL. The amount that might escape would not necessarily be anymore than might escape as the door is cycled to evacuate personnel. What does escape will be filtered by the Primary Plant Ventilation System, the same as if the accident were to occur in the fuel building. In summary, not only is the accident clearly bounded by the existing analysis, the actual increase in release of radioactive material outside the plant will be insignificant if there is any measurable increase at all.

Based on the above, allowing the PAL doors to remain open during movement of irradiated fuel and core alterations, has no significant effect on the probability or consequences of an accident previously evaluated.

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

The change does not add new hardware. The only change in the operation of the plant is that the PAL doors will remain open during movement of irradiated fuel and core alterations. Because the current fuel handling accident analysis considers fuel handling accidents in either the Fuel Building or the Containment Building, the current fuel handling accident analysis remains bounding for the proposed change. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The assumptions used to calculate the offsite dose resulting from a fuel handling accident in [the] Containment Building are equivalent to assuming that the PAL remains open for the entire accident and that no filtration occurs. Since no credit was taken for any containment barrier or ventilation system filtration, the dose to the public as calculated in the analysis is not affected by this change. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: November 21, 1995

Brief description of amendments: The proposed amendment would revise the core safety limit curves and revised N-16 Overtemperature reactor trip setpoints as a result of the reload analyses for CPSES Unit 2, Cycle 3. In addition, the minimum required Reactor Coolant System (RCS) flow is increased and an administrative enhancement is included in the footnotes of the RCS flow - low reactor trip function setpoint for both Units.

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

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

A. Increase in Unit 2 minimum required flow

This revision increases the Unit 2 minimum required RCS flow rate assumed in the safety analyses by 3.6%. The actual core flow is unchanged and is approximately 6.6% higher than the value assumed in previous accident analyses. The remaining 3.0% flow is sufficient to account for all uncertainties associated with the core flow measurement.

Since this change only involves analysis methodology and does not affect the actual core flow, it does not increase the actual probability or consequences of any postulated accident.

When considered separately, increasing the minimum required RCS flow is a conservative change. Although there is no impact on the initiation of any postulated accidents, the potential severity of the affected accidents is typically less when flow is increased. In general, the increased ability to remove heat from the fuel will reduce the peak temperature seen by the fuel and reduce the potential for undesirable boiling conditions. Thus, the increase in the assumed RCS flow will not increase the probability or consequences of an accident previously analyzed.

B. Revision to the Unit 2 Core Safety Limits

Analyses of reactor core safety limits are required as part of reload calculations for each cycle. TU Electric has performed in-house analyses of the Unit 2, Cycle 3 core to determine the reactor core safety limits. The newer methodologies and safety analysis values result in new operating curves which, in general, permit plant operation over a similar range of acceptable conditions. This change means that if a transient were to occur with the plant operating at the limits of the new curve, a higher temperature and power level might be attained than if the plant were operating within the bounds of the old curves. However, since the new curves were developed using approved methodologies which are wholly consistent with and do not represent a change in the Technical Specification bases for safety limits, all applicable postulated transients will continue to be properly mitigated. As a result, there will be no significant increase in the consequences, as determined by accident analyses, of any accident previously evaluated.

C. Revision to Unit 2 Overtemperature N-16 Reactor Trip Setpoints, Parameters and Coefficients

As a result of changes discussed, the Overtemperature N-16 reactor trip setpoint has been recalculated. These trip setpoints help ensure that the core safety limits are maintained and that all applicable limits of the safety analysis are met.

Based on the calculations performed, the safety analysis value for Overtemperature N-16 reactor trip setpoint has changed. This essentially means if a transient were to occur, the actual temperature and power level could be slightly higher. However, the analyses performed show that, using the TU Electric methodologies, all reactor core safety limits are met and all applicable limits of the safety analysis are met. This parameter has a setpoint which allows the mitigation of postulated accidents and has no impact on accident initiation. Therefore, the changes in safety analysis values do not involve an increase in the probability of an accident and, based on satisfying the core safety limits and all applicable safety analysis limits, there is no significant increase in the consequences of any accident previously evaluated.

In addition, the changes result in setpoint values which potentially offer safety benefits. The risk of turbine runbacks or reactor trips due to upper plenum flow anomalies will be minimized with a higher overtemperature setpoint, thus reducing potential challenges to the plant safety systems. A final benefit is that the new methods for considering N-16 setpoints and values will be consistent with Unit 1, which reduces the potential for personnel error due to unit differences.

Considering both the safety analysis impact and the benefits described above, the changes in N-16 setpoints and parameters will result in slight reduction in the probability of an accident and do not significantly increase the consequences of an accident previously evaluated.

D. Deletion of footnotes associated with the RCS flow - low reactor trip setpoint

In lieu of revising the footnotes to support the Unit 2 Cycle 3 operation, the deletion of the footnote is proposed. Further, for consistency with Unit 2, the same change is proposed for Unit 1. This change will not affect current plant practice; however, it will impose a more restrictive RCS flow - low setpoint than is currently required. The RCS flow - low reactor trip setpoint is currently specified in Technical Specification Table 2.2-1, Functional Unit 12.b, to be 90% of the minimum measured RCS flow. The proposed change would require the setpoint to be 90% of the instrument span where 100% of instrument span approximately corresponds to the actual RCS flow. The actual RCS flow is verified to be greater than the RCS flow assumed in the accident analysis through compliance with Technical Specification 3.2.5. Thus, through deletion of the footnotes, the RCS volumetric flow corresponding to the reactor trip setpoint will be greater than or equal to the volumetric flow allowed by the current specifications.

In summary, the proposed deletion of the footnotes will have no impact on current plant operations. A possible relaxation of the RCS flow - low setpoint which is currently allowed by the Technical Specifications will be removed without creating the potential for unnecessary plant trips.

The RCS flow - low reactor trip setpoint can have no effect on the probability of an accident. Because the reactor will be tripped at or prior to the conditions assumed in the accident analyses, there will be no effect on the consequences of an accident previously identified.

SUMMARY

The changes in the amendment request applies new NRC approved methodologies, changes in safety analysis values, new core safety limits and new N-16 setpoint and parameter values to assure that all applicable safety analysis limits have been met. The potential for an operational transient to occur has been reduced and there has been no significant impact on the consequences of any accident previously evaluated.

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

The proposed changes involve the use of revised safety analysis values and the calculation of new reactor core safety limits and reactor trip setpoints. As such, the changes play an important role in the analysis of postulated accidents but none of the changes effect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

In reviewing and approving the methods used for safety analyses and calculations, the NRC has approved the safety analysis limits which establish the margin of safety to be maintained. While the actual impact on safety is discussed in response to question 1, the impact on margin of safety is discussed below.

A. Increase in the Unit 2 minimum required flow

In performing the DNB-related analyses, the Reactor Coolant System flow rate assumed in these analyses is increased by 3.6 percent to insure that all applicable limits of the safety analysis are met. The Technical Specification 3/4.2.5 limit for this parameter will be changed to insure that it is maintained within the normal steady-state envelope of operation assumed in the transient and accident safety analyses (i.e., ensuring that the RCS flow rate assumed in the safety analyses remains valid). The Technical Specification limits are consistent with the initial safety analysis assumption (plus uncertainties) and have been analytically demonstrated to be adequate to maintain a minimum DNBR at or above the safety analysis DNBR limit throughout each analyzed transient. Because the 95/95 DNBR acceptance criteria is met with the proposed change and assumptions of the safety analyses are maintained valid by the Technical Specification limits, there is no change in a margin of safety.

B. Revision to the Unit 2 Reactor Core Safety Limits

The TU Electric reload analysis methods have been used to determine new reactor core safety limits. All applicable safety analysis limits have been met. The methods used are wholly consistent with Technical Specification BASES 2.1 which is the bases for the safety limits. In particular, the curves assure that for Unit 2, Cycle 3, the calculated DNBR is no less than the safety analysis limit and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

In conjunction with the reactor core safety limit methodology, the NRC approved TUE-1 DNB correlation is used for performing DNB-related analyses. This correlation will be applied to the core configuration of CPSES Unit 2, Cycle 3 and future core configurations. The TUE-1 correlation DNBR limit is established such that there is a 95 percent probability with 95 percent confidence level that DNB will not occur when the minimum DNBR for the limiting fuel is greater than or equal to the TUE-1 correlation DNBR limit. This 95/95 criteria defines the ''margin of safety'' for the DNB-related analysis and remains valid even though the DNB correlation and associated correlation limit are changed. Margin is provided in the DNB-related analysis for known and potential effects such as hydraulic differences between the two co-resident fuel assembly designs and the presence of the Reactor Coolant System lower plenum flow anomaly. The TUE-1 correlation DNBR limit plus margin constitutes the safety analysis DNBR limit. The accident analyses are performed to ensure that the safety analysis DNBR limit acceptance criteria are satisfied. Because the 95/95 DNBR acceptance criteria remains valid and continues to be satisfied, no change in a margin of safety occurs.

C. Revision to Unit 2 Overtemperature N-16 Reactor Trip Setpoints, Parameters and Coefficients

Because the reactor core safety limits for CPSES Unit 2, Cycle 3 are recalculated, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint which protect the reactor core safety limits must also be recalculated. The Overtemperature N-16 reactor trip setpoint helps prevent the core and Reactor Coolant System from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. The most relevant design basis analysis in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) which is affected by the change in the safety analysis value for the CPSES Unit 2 Overtemperature N-16 reactor trip setpoint is the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2). This event has been re-analyzed with the revised safety analysis value for the Overtemperature N-16 reactor trip setpoint to demonstrate compliance with event specific acceptance criteria. Because all event acceptance criteria are satisfied, there is no degradation in a margin of safety.

The nominal Reactor Trip System instrumentation setpoints values for the Overtemperature N-16 reactor trip setpoint (Technical Specification Table 2.2-1) are determined based on a statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. The total uncertainty plus additional margin is applied in a conservative direction to the safety analysis trip setpoint value to arrive at the nominal and allowable values presented in Technical Specification Table 2.2-1. Meeting the requirements of Technical Specification Table 2.2-1 assures that the Overtemperature N-16 reactor trip setpoint assumed in the safety analyses remains valid. The CPSES Unit 2, Cycle 3 Overtemperature N-16 reactor trip setpoint is different from previous cycles which provides more operational flexibility to withstand mild transients without initiating automatic protective actions. Although the setpoint is different, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint are consistent with the safety analysis assumption which has been analytically demonstrated to be adequate to meet the applicable event acceptance criteria. Thus, there is no reduction in a margin of safety.

D. Deletion of footnotes associated with the RCS flow - low reactor trip function

The deletion of the footnotes, and the potential relaxation of the RCS flow - low setpoint which could be used, will provide further assurance that, in the event of a partial loss of forced RCS flow or locked rotor transient, a reactor trip signal would be initiated prior to the conditions assumed in the accident analyses. Thus, the accident analyses are unaffected, and there is no reduction in a margin of safety.

SUMMARY

The proposed changes to the CPSES Technical Specifications involve using NRC-approved licensing analysis methods developed by TU Electric to determine the Technical Specification reactor core safety limits and perform DNB-related analysis for CPSES Unit 2, Cycle 3. The DNB-related analyses are performed by TU Electric using a qualified, state-of-the-art departure from nucleate boiling (DNB) correlation, TUE-1, which has also been approved by the NRC for the CPSES Unit 2, Cycle 3 core configuration. In performing these analyses, the minimum required Reactor Coolant System flow rate is increased by 3.6 percent. Because the core safety limits for CPSES Unit 2, Cycle 3 are recalculated, the Reactor Trip System instrumentation setpoints values for the Overtemperature N-16 reactor trip setpoint which protect the core safety limits are also recalculated.

Using the NRC approved TU Electric methods, the reactor core safety limits are determined such that all applicable limits of the safety analyses are met, particularly the 95/95 DNBR limit. The Technical Specification 3/4.2.5 limits for the DNB Parameters insure the assumptions in the safety analyses remain valid. Because the applicable event acceptance criteriacontinue to be met, there is no 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: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019

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NRC Project Director: William D. Beckner

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

Date of amendment request: October 17, 1995

Description of amendment request: The proposed amendment would modify the North Anna Power Station, Units 1 and 2 Technical Specifications (TS) to allow both of the containment personnel airlock doors to remain open during refueling operations, delete the license condition referencing the analyses for limiting doses to the control room operators, and modify the TS Bases to clarify the emergency power system requirements relative to mitigation of the consequences of a Fuel Handling Accident.

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:

There is no significant change in the probability or consequences of an accident previously evaluated. There are no system changes which would increase the probability of an accident occurring. Allowing both personnel airlock doors to remain open during core alterations or fuel movement inside containment will not have any impact on the probability of a Fuel Handling Accident either in containment or in the fuel building. The consequences of a Fuel Handling Accident have been investigated by performing a reanalysis with no credit for isolation or filtration by the Fuel Building or containment ventilation systems. The Exclusion Area Boundary [EAB] and Low Population Zone [LPZ] doses for a Fuel Handling Accident without credit for iodine filtration remain well within (<25%) of the NRC regulatory limits of 10 CFR [Part] 100. The predicted control room operator doses remain bounded by the limiting case for control room doses and within the regulatory limits of General Design Criterion [GDC] 19. In addition, the action to clarify the responses to NRC question 6.72 [of the original Final Safety Analysis Report] will not increase the probability or consequences of the Fuel Handling Accident.

No new accident types or equipment malfunction scenarios are introduced as a result of the clarification to the Virginia Power response to [NRC question] 6.72 or as a result of these changes in analysis methods or the proposed Technical Specifications changes to allow both personnel airlock doors to remain open during core alterations or fuel movement inside containment. Therefore, there is no possibility of an accident of a different type than any previously evaluated in the North Anna USFAR [Updated Final Safety Analysis Report].

There is no significnt reduction in the margin of safety. An evaluation of the Fuel Handling Accident doses at the EAB, the LPZ and to control room operators has been performedand it has been concluded that the acceptance criteria defined by GDC-19, 10 CFR 100, and the NRC Standard Review Plan will continue to be met.

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

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23212.

NRC Project Director: David B. Matthews

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

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

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

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

Date of application for amendments: September 13, 1995, as amended November 27, 1995

Brief description of amendments: The proposed amendments would permit the licensee to implement the performance-based option provided by 10 CFR Part 50, Appendix J, which allows leakage testing intervals to be based on system and component testing performance.

Date of publication of individual notice in Federal Register: December 12, 1995 (60 FR 63739)

Expiration date of individual notice: January 11, 1996

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

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.

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

Date of application for amendments: October 3, 1995

Brief description of amendments: The amendments delete Sections 2.B.(7)(a) and (b) of

Facility Operating License No. NPF-51 (Unit 2) and Sections 2.b.(6)(a) and (b) of

Facility Operating License No. NPF-74 (Unit 3) relating to certain previous sale and leaseback transactions that were added by Amendment No. 3 for NPF-51 and Amendment No. 1 for NPF-74.

Date of issuance: December 8, 1995

Effective date: December 8, 1995

Amendment Nos.: Unit 2 - Amendment No. 91; Unit 3 - Amendment No. 74

Facility Operating License Nos. NPF-51 and NPF-74: The amendments revised the license.

Date of initial notice in Federal Register: November 8, 1995 (60 FR 56363) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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

Date of application for amendments: October 23, 1995

Brief description of amendments: The amendments revised the Technical Specifications to delete the applicability of the primary coolant water chemistry limits when the primary system is being chemically decontaminated and the reactor vessel is defueled.

Date of issuance: December 13, 1995

Effective date: December 13, 1995

Amendment Nos.: 180 and 211

Facility Operating License Nos. DPR-71 and DPR-62.

Date of initial notice in Federal Register: The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 1995. 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.

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: September 14, 1995, as supplemented November 8, 1995.

Brief description of amendments: The amendments allow the use of an alternate zirconium based fuel cladding, ZIRLO, and permit limited substitution of fuel rods with ZIRLO filler rods. In addition, a clarification and an editorial change have been included.

Date of issuance: December 19, 1995

Effective date: December 19, 1995

Amendment Nos.: 78 and 70

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

Date of initial notice in Federal Register: October 25, 1995 (60 FR 54716) The November 8, 1995 letter, provided clarifying information that did not change the scope of the September 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 December 19, 1995No significant hazards consideration comments received: No

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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 amendments: September 15, 1995.

Brief description of amendments: The amendments upgrade the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG-0123, ''Standard Technical Specification General Electric Plants BWR/4.'' The application dated September 15, 1995, contains some of the TSUP open items from previous Dresden and Quad Cities TS amendments issued by the NRC.

Date of issuance: December 19, 1995Effective date: Immediately, to be implemented no later than June 30, 1996.

Amendment Nos.: 145, 139, 167 and 163

Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 5, 1995 (60 FR 52220) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 1995.No significant hazards consideration comments received: No

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.

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: December 8, 1995

Brief description of amendment: The amendment revises Technical Specification (TS) 3.1.A.5 to revise the wording to allow a single train of Power-Operated Relief Valves (PORVs)/Block Valves to be closed and deenergized indefinitely. The proposed change is administrative and is intended to correct inconsistencies between the intended operation of the PORVs/Block Valves and the language of the TSs.

Date of issuance: December 8, 1995

Effective date: As of the date of issuance to be implemented immediately.

Amendment No.: 185

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: NoThe Commission's related evaluation of the amendment, emergency circumstances and consultation with the State, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated December 8, 1995.

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: Ledyard B. Marsh

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: October 31, 1994

Brief description of amendments: The amendments remove the stroke times for the steam generator power operated relief valves from Technical Specification Tables 3.6-2a and 3.6-2b.

Date of issuance: December 18, 1995

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

Amendment Nos.: Unit 1 - 139 - Unit 2 - 133

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

Date of initial notice in Federal Register: February 15, 1995 (60 FR 8745) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 18, 1995.No significant hazards consideration comments received: No

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

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: September 4, 1993, as supplemented by letters dated February 16, 1994, and August 4, 1995

Brief description of amendments: The license amendments revised the Arkansas Nuclear One Industrial Security Plan.

Date of issuance: December 19, 1995

Effective date: December 19, 1995

Amendment Nos.: 183 and 172

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

Date of initial notice in Federal Register: November 8, 1995 (60 FR 56368) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 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 6, 1993, as supplemented by letters dated May 12, August 9, and September 18, 1995.

Brief description of amendment: The amendment changes the Appendix A TSs to allow installation of steam generator tube repair sleeves at the Waterford Steam Electric Station, Unit 3. The sleeves are designed and manufactured by Combustion Engineering Incorporated.

Date of issuance: December 14, 1995

Effective date: December 14, 1995, to be implemented within 60 days

Amendment No.: 117

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

Date of initial notice in Federal Register: January 19, 1994 (59 FR 2868) The May 12, August 9, and September 18, 1995, letters provided additional 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 December 14, 1995.No significant hazards consideration comments received: No

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

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

Date of amendment request: September 7, 1993, as supplemented by letters dated February 8, 1994, and August 9, 1995.

Brief description of amendment: The amendment revised the license condition on physical security and approves the revision to Physical Security Plan for the Waterford Steam Electric Station, Unit 3.

Date of issuance: December 19, 1995

Effective date: December 19, 1995

Amendment No.: 118

Facility Operating License No. NPF-38. Amendment revised the license. The additional information contained in the supplemented letter dated August 9, 1995, was 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.

Date of initial notice in Federal Register: March 30, 1994 (59 FR 14887) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 19, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 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: July 26, 1995

Brief description of amendments: The amendments consist of changes to the Technical Specifications relating to nuclear instrumentation system adjustments based on calorimetric measurements at reduced power levels.Date of issuance: December 12, 1995

Effective date: December 12, 1995

Amendment Nos. 180 and 174Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47617) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 12, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendment: January 26, 1995, as supplemented March 9 and May 24, 1995

Brief description of amendment: This amendment increases the allowable U-235 enrichment of fuel to be stored in the new and spent fuel storage facilities.

Date of issuance: December 15, 1995

Effective date: December 15, 1995

Amendment No.: 151

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

Date of initial notice in Federal Register: April 26, 1995 (60 FR 20517) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: October 24, 1995

Brief description of amendment: The amendment revised the Technical Specifications to reflect the approval for the River Bend Station to use 10 CFR Part 50, Appendix J, Option B for the containment leak rate testing.

Date of issuance: December 19, 1995

Effective date: December 19, 1995

Amendment No.: 84

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

Date of initial notice in Federal Register: November 8, 1995 (60 FR 56368) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 19, 1995.No significant hazards consideration comments received. No.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

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: May 25, 1995 (AEP:NRC:1200B)

Brief description of amendments: The amendments change the surveillance frequency for the manual actuation function for main steam line isolation from monthly to quarterly and delete obsolete footnotes associated with previous surveillance interval extensions from Unit 2 Table 4.3-2.

Date of issuance: December 13, 1995

Effective date: December 13, 1995, with full implementation within 45 days

Amendment Nos.: 204 and 189

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

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35081)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 1995. No significant hazards consideration comments received: No.

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

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: August 8, 1995

Brief description of amendment: This amendment modifies the definitions of Transthermal (Condition 4), Hot Shutdown (Condition 5), and Hot Standby (Condition 6) reactor operating conditions. The Transthermal and Hot Shutdown Conditions are modified to establish an applicable range of subcriticality and be consistent with other Definitions. The wording of Hot Standby is modified to remove reference to control rod position, consistent with NUREG-1432, Standard Technical Specifications for Combustion Engineering Plants, Revision 1, dated April 1995.

Date of issuance: December 15, 1995

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

Amendment No.: 154

Facility Operating License No. DPR-36: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52931) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

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

Date of amendment request: June 27, 1995

Brief description of amendment: The amendment revises Technical Specification (TS) 2.2 on chemical and volume control system (CVCS) to reformat and clarify the requirements and make them more consistent with the requirements of the Combustion Engineering Standard Technical Specifications (STS), as presented in NUREG-0212, Revision 2.

Date of issuance: December 12, 1995

Effective date: December 12, 1995

Amendment No.: 171

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39447) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

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 application for amendments: February 10, 1995, as supplemented by letter dated November 10, 1995.

Brief description of amendments: These amendments (1) modify the Susquehanna Steam Electric Station, Unit 1 and 2 Technical Specifications to extend the allowable out-of-service times (AOTs) for maintenance and repair and the surveillance test intervals (STIs) between channel functional tests for the following groups of instruments: reactor protection systems instrumentation (TS 3.3.1), isolation actuation instrumentation (TS 3.3.2), emergency core cooling system actuation instrumentation (TS 3.3.3), ATWS (anticipated transient without scram) recirculation pump trip system instrumentation (TS 3.3.4.1), end-of-cycle recirculation pump trip system instrumentation (TS 3.3.4.2), reactor core isolation cooling system (RCIC) actuation instrumentation (TS 3.3.5), control rod block instrumentation (TS 3.3.6), radiation monitoring instrumentation (TS 3.3.7.1), and feedwater/main turbine trip system actuation instrumentation (TS 3.3.90); (2) change the required actions and AOTs for the instruments listed above to make requirements consistent with supporting analysis in General Electric topical reports and change additional actions required to prevent extended AOTs from resulting in extended loss of instrument function; (3) change the required actions and AOTs for the instruments listed above for instrumentation associated with the ADS (automatic depressurization system), recirculation pump trip, and pump suction lineup for HPCI (high pressure core injection) and RCIC; (4) change applicability requirements and required actions for the reactor vessel water level-low, level 3 function that isolates the RHR (residual heat removal) system shutdown cooling system so that the function is required to be operable in operational conditions 3,4, and 5 to prevent inadvertent loss of reactor coolant via the RHR shutdown cooling system; (5) remove notes in Table 3.3.2-1, 3.3.2-2, and 4.3.1-1 related to maintenance on leak detection temperature detectors and remove the note toTS 3.3.6 for Unit 1 related to a previous relief from TS 3.0.4; and (6) reformat, renumber, and/or reword existing requirements to incorporate the changes listed above.

Date of issuance: December 18, 1995

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

Amendment Nos.: 155 and 126

Facility Operating License Nos. NPF-14 and NPF-22. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 29, 1995 (60 FR 16194) The supplemental letter provided corrected TSs and did not change the original proposed no significant hazards consideration nor the Federal Register notice.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 18, 1995No significant hazards consideration comments received: No

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

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

Date of amendments request: September 26, 1995

Brief description of amendments: The amendments change the containment air lock door seal leakage rate from ''no detectable seal leakage'' to ''less than or equal to 0.01 L_a'' when the gap between the door seals is pressurized to greater than or equal to 10 psig for a period of not less than 15 minutes.

Date of issuance: December 8, 1995

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

Amendment Nos.: 118 and 109

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

Date of initial notice in Federal Register: November 8, 1995 (60 FR 56370) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

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: August 7, 1995 (TS 95-17)

Brief description of amendments: The changes relocate the heat flux hot channel factor penalty from Surveillance Requirement 4.2.2.2.e.1 to the Core Operating Limits Report and replace the methodology (WCAP-10216-P-A) listed in Technical Specification 6.9.1.14.a.2 with WCAP-10216-P-A, Revision 1A.

Date of issuance: December 11, 1995

Effective date: December 11, 1995

Amendment Nos.: 216 and 206

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

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45186) The Commission's related evaluation of the amendment is contained in a Safety

Evaluation dated December 11, 1995.No significant hazards consideration comments received: None

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402No significant hazards consideration comments received: None

TU Electric Company, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: August 15, 1995 (TXX-95215)

Brief description of amendments: These changes relocated the Shutdown Margin limits from the Technical Specifications (TSs) to the Core Operating Limits Report (COLR). The changes were consistent with the intent of Generic Letter 88-16 which provides guidelines for the removal of cycle-specific parameter limits from the TSs.

Date of issuance: December 15, 1995

Effective date: December 15, 1995

Amendment Nos.: Unit 1 - Amendment No. 44; Unit 2 - Amendment No. 30

Facility Operating License Nos. NPF-87 and NPF-89. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52935) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 15, 1995.No significant hazards consideration comments received: No

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

Union Electric Company, Docket No. 50-483, Callaway Plant, Callaway County, Missouri

Date of amendment request: April 26, 1995

Brief description of amendment: The amendment revises Technical Specification (TS) 3/4.7.6 to reduce the upper limit on the flow rate through the control room filtration subsystem and adopts ASTM D-3803-1989 as the laboratory testing standard for control room filtration and control building pressurization charcoal adsorber. The amendment also revises the Bases for TS 3/4.7.6 to reflect the changes.

Date of issuance: December 20, 1995

Effective date: December 20, 1995, to be implemented within 30 days from the date of issuance.

Amendment No.: 106

Facility Operating License No. NPF-30. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 23, 1995 (60 FR 27345) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 20, 1995.No significant hazards consideration comments received: No.

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

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: June 14, 1995, as supplemented by letters dated July 13, 1995, and August 22, 1995.I11*Brief description of amendment:* The amendment revises Technical Specification (TS) 3.2.3, ''Nuclear Enthalpy Rise Hot Channel Factor,'' TS 6.9.1.9, ''Core Operating Limits Report,'' and the associated Bases sections. The revisions incorporate changes associated with the planned implementation of advanced nuclear and core thermal-hydraulic design methodologies licensed from Westinghouse Electric Corporation for core reload design, starting with Cycle 9.

Date of issuance: December 8, 1995

Effective date: December 8, 1995, to be implemented prior to restart from the eighth refueling outage, which is scheduled to begin in March 1996.

Amendment No.: 92

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39456) The August 22, 1995, supplemental letter forwarded the nonproprietary version of Wolf Creek Nuclear Operating Corporation's safety evaluation and analysis provided in the June 14, 1995, submittal and did not change the staff's original no significant hazards determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 8, 1995.No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: August 22, 1995

Brief description of amendment: The amendment revises the requirements of Technical Specification (TS) 3.3.1 and TS 3.3.2 and relocate Tables 3.3-2 and 3.3-5 and applicable Bases, which provide the response time limits for the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS) instruments, from the TS to the Updated Safety Analysis Report (USAR). The licensee has stated that the next USAR change request will include these changes.

Date of issuance: December 12, 1995

Effective date: December 12, 1995, to be implemented within 60 days of issuance.

Amendment No.: 93

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49950) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 1995.No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Dated at Rockville, Maryland, this 21st Day of December 1995.

For the Nuclear Regulatory Commission

Steven A. Varga,

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

[Doc. 96–1 Filed 1–2–96; 8:45 am]

BILLING CODE 7590–O1–F

[Docket Nos. 50–295 and 50–304]

Commonwealth Edison Company, (Zion Nuclear Power Station, Unit Nos. 1 and 2); Exemption

I

The Commonwealth Edison Company (ComEd, the licensee) is the holder of Facility Operating License Nos. DPR–39 and DPR–48, which authorize operation of the Zion Nuclear Power Station, Units 1 and 2 (the facilities). The licenses provide, among other things, that the facilities are subject to all the rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (the Commission) now or hereafter in effect.