Division, Immigration and Naturalization Service.

(4) *Affected public who will be asked or required to respond, as well as a brief abstract:* Primary: Individuals or Households. This form is used by the INS to request a verification of the military or naval service claim by an applicant filing for naturalization on the basis of honorable service in the United States Armed Forces.

(5) *An estimate of the total number of respondents and the amount of time estimated for an average respondent to respond:* 45,000 respondents at 10 minutes (.166) hours per response.

(6) *An estimate of the total public burden (in hours) associated with the collection:* 7,470 annual burden hours.

If additional information is required during the first 60 days of this same regular review period contact Mr. Robert B. Briggs, Clearance Officer, United States Department of Justice, Information Management and Security Staff, Justice Management Division, Suite 850, Washington Center, 1001 G Street, NW., Washington, DC 20530.

Dated: July 24, 1997.

Robert B. Briggs,

Department Clearance Officer, United States Department of Justice.

[FR Doc. 97–20003 Filed 7–29–97; 8:45 am]

BILLING CODE 4410–18–M

DEPARTMENT OF JUSTICE

Immigration and Naturalization Service

Agency Information Collection Activities: Proposed Collection; Comment Request

ACTION: Request OMB emergency approval; Application for travel document.

The Department of Justice, Immigration and Naturalization Service (INS) has submitted the following information collection request (ICR) utilizing emergency review procedures, to the Office of Management and Budget (OMB) for review and clearance in accordance with the section 1320.13(a)(2)(iii) of the Paperwork Reduction Act of 1995. The INS has determined that it cannot reasonably comply with the normal clearance procedures under this part because normal clearance procedures are reasonably likely to prevent or disrupt the collection of information. This information collection is needed prior to the expiration of established time periods. OMB approval has been requested by July 31, 1997. If granted, the emergency approval is only valid for 90 days. All comments and/or questions pertaining to this pending request for emergency approval must be directed to OMB, Office of Information and Regulatory Affairs, Attention: Ms. Debra Bond, 202–395–7316, Department of Justice Desk Officer, Washington, DC 20503. Comments regarding the emergency submission of this information collection may also be telefaxed to Ms. Bond at 202–395–6974.

During the first 60 days of this same period, a regular review of this information collection is also being undertaken. During the regular review period, the INS requests written comments and suggestions from the public and affected agencies concerning the proposed collection of information. Comments are encouraged and will be accepted until September 29, 1997. During the 60-day regular review all comments and suggestions, or questions regarding additional information, to include obtaining a copy of the proposed information collection instrument with instructions, should be directed to Mr. Richard A. Sloan, 202–514–3291, Director, Policy Directives and Instructions Branch, Immigration and Naturalization Service, U.S. Department of Justice, Room 5307, 425 I Street NW., Washington, DC 20536. Your comments should address one or more of the following four points.

(1) Evaluate whether the proposed collection of information is necessary for the proper performance of the functions of the agency's, including whether the information will have practical utility;

(2) Evaluate the accuracy of the agencies estimate of the burden of the proposed collection of information, including the validity of the methodology and assumptions used;

(3) Enhance the quality, utility, and clarity of the information to be collected; and

(4) Minimize the burden of the collection of information on those who are to respond, including through the use of appropriate automated, electronic, mechanical, or other technological collection techniques or other forms of information technology, e.g., permitting electronic submission of responses.

Overview of this information collection:

(1) *Type of Information Collection:* Extensionof a currently approved information collection.

(2) *Title of the Form/Collection:* Application for Travel Document.

(3) *Agency form number, if any, and the applicable component of the Department of Justice sponsoring the collection:* Form I–131. Adjudications Division, Immigration and Naturalization Service.

(4) *Affected public who will be asked or required to respond, as well as a brief abstract:* Primary: Individuals or Households. This form is used by permanent or conditional residents, refugees or asylees and aliens abroad seeking to apply for a travel document to lawfully reenter the United States or be paroled for humanitarian purposes into the United States.

(5) *An estimate of the total number of respondents and the amount of time estimated for an average respondent to respond:* 335,000 respondents at 55 minutes (.90) hours per response.

(6) *An estimate of the total public burden (in hours) associated with the collection:* 301,500 annual burden hours.

If additional information is required during the first 60 days of this same regular review period contact Mr. Robert B. Briggs, Clearance Officer, United States Department of Justice, Information Management and Security Staff, Justice Management Division, Suite 850, Washington Center, 1001 G Street NW., Washington, DC 20530.

Dated: July 24, 1997.

Robert B. Briggs,

Department Clearance Officer, United States Department of Justice.

[FR Doc. 97–20004 Filed 7–29–97; 8:45 am]

BILLING CODE 4410–18–M

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 3, 1997, through July 18, 1997. The last biweekly notice was published on July 16, 1997.

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

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

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

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

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

By August 29, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

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

Date of amendments request: May 23, 1997

Description of amendments request: The proposed amendment would revise Technical Specification 3/4.4.4 to allow the installation of ABB/CE welded sleeves, in accordance with ABB/CE Topical Report CEN-630-P, ''Repair of 3/4 Inch Outer Diameter Steam Generator Tubes Using Leak Tight Sleeves,'' Revision 1, in the Palo Verde Units 1, 2 and 3 steam generators.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below: 1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Analytical verifications were performed using design and operating transient parameters selected to envelope loads imposed during normal operating and accident conditions. Fatigue and stress analysis of sleeved tube assemblies were completed in accordance with the requirements of Section III of the ASME Code. The results of qualification testing, analysis and plant operating experience at other facilities demonstrates that the sleeving process is an acceptable means of maintaining steam generator tube integrity. The sleeve configuration has been designed and analyzed in accordance with the structural margins specified in Regulatory Guide 1.121 (RG 1.121). Furthermore, the installed sleeve will be monitored through periodic inspections on a sample basis with eddy current techniques. A sleeve-specific plugging margin, per the recommendations of Regulatory Guide 1.121, has been specified with appropriate allowances for NDE uncertainty and defect growth rate. Therefore, since the sleeve provides the same protection against a tube rupture as the original tube, the use of sleeves does not involve a significant increase in the probability of an accident previously evaluated.

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

The consequences of accidents previously analyzed are not increased as a result of sleeving activities. The hypothetical failure of the sleeve would be bounded by the current steam generator tube rupture analysis contained in the PVNGS UFSAR. Due to the slight reduction in diameter caused by the sleeve wall thickness, it is expected that the primary release rates would be less than assumed for the steam generator tube rupture analysis, and, therefore, would result in lower primary fluid mass release to the secondary system. Additionally, further conservatism is introduced if the break were postulated to occur at a location on the tube higher than the location where a sleeve is installed. The overall effect would be reduced steam generator tube rupture release rates. The minimal reduction in flow area associated with a tube sleeve has no significant affect on steam generator performance with respect to heat transfer or system flow resistance and pressure drop. The installation of sleeves rather than plugging also maintains a greater heat transfer surface in the steam generator. In any case, the impacts are bounded by evaluations which demonstrate the acceptability of tube plugging, which totally removes the tube from service.

Therefore, in comparison to plugging, tube sleeving is considered a significant improvement with respect to steam generator performance. Therefore, based on the above, the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated.

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

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

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

ABB Combustion Engineering, Inc., Report CEN-630-P, Revision 01, ''Repair of 3/4'' O.D. Steam Generator Tubes Using Leak Tight Sleeves'' dated November, 1996, demonstrates that the repair of degraded steam generator tubes using tube sleeves will result in tube bundle integrity consistent with the original design basis. Extensive analyses and testing have been performed on the sleeve and sleeve to tube joints to demonstrate that the design criteria are met. The proposed amendments have no significant effect on the configuration of the plant, and the change does not affect the way in which the plant is operated. Therefore, reactor operation with sleeves installed in the steam generator tubes does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Steam generator tube integrity is maintained under the same limits for sleeved tubes as for unsleeved tubes, ie., Section III of the ASME Code and Regulatory Guide 1.121. The portions of the installed sleeve assembly which represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. The degradation limit at which a sleeve/tube boundary is considered inoperable has been analyzed in accordance with Regulatory Guide 1.121 and is specified in the proposed amendment. Eddy current detectability of flaws has been verified by ABB Combustion Engineering. Additionally, the Technical Specifications continue to require monitoring and restriction of primary- to- secondary system leakage through the steam generators. The minimal reduction in RCS flow due to sleeving results in an insignificant impact on RCS operation during normal or accident conditions and is bounded by tube plugging evaluations.

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

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

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

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

NRC Project Director: William H. Bateman

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 amendments request: 3April 30, 1997

Description of amendments request: The proposed amendments would revise Surveillance Requirements (SRs) 4.7.2.b.2 and 4.7.2.c in the Technical Specifications for the Brunswick Steam Electric Plant, Units 1 and 2. These SRs require periodic testing of the control room emergency ventilation system charcoal filters. The proposed amendments would revise the temperature and relative humidity conditions under which the testing is performed. The revised conditions were selected to approximate operating or accident conditions. Testing at the revised conditions is more conservative than testing at the currently required conditions. Additionally, the proposed amendments would relax the acceptance criterion for filtration efficiency from 95% to a value corresponding to a filtration efficiency of 90%. The 90% value is the filtration efficiency assumed in the current bounding calculations for control room dose under accident conditions.

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

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

The proposed amendments revise Surveillance Requirements 4.7.2.b.2 and 4.7.2.c to require testing of the control room emergency ventilation system (CREVS) charcoal in accordance with ASTM D3803-1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.'' Currently, Surveillance Requirements 4.7.2.b.2 and 4.7.2.c to [sic] require testing in accordance with the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, 1976. The purpose of the CREVS is to mitigate an accident. It is not associated with any initiating events and, therefore, cannot affect the probability of any accident.

ASTM D3803-1989 is an industry accepted standard for charcoal filter testing. The conditions employed by this standard were selected to approximate operating or accident conditions of a nuclear reactor which would severely reduce the performance of activated carbons. The ASTM D3803-1989 testing is more stringent than that required by the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, 1976. Specifically, the testing temperature of ASTM D3803-1989 is 30.0 [plus or minus] 0.2°C versus 80°C for the Regulatory Guide 1.52 testing. Also, ASTM D3803-1989 requires a relative humidity of 93 to 96% versus [greater than or equal to] 70% for the Regulatory Guide 1.52 testing. Both these parameters result in the ASTM D3803-1989 test being a more conservative test [than] that required by the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, 1976.

The proposed changes to Surveillance Requirements 4.7.2.b.2 and 4.7.2.c require that charcoal samples tested in accordance with the methodology of ASTM D3803-1989 meet the acceptance criteria of < 5.0% penetration of methyl iodide. This corresponds to a 90% filtration efficiency which is the filtration efficiency assumed in the current bounding calculations of control room doses. As such, the proposed acceptance criteria of < 5.0% penetration of methyl iodide ensures that General Design Criterion 19 dose limits for control room operators are not exceeded.

Therefore, the proposed amendments do not involve an increase in the consequences of an accident.

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

As stated above, the proposed amendments revise the required testing methodology for the CREVS charcoal. The CREVS is not associated with any initiating events. The system design is not affected by the proposed change. Therefore, the proposed amendments cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments upgrade the CREVS charcoal testing requirements from the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, 1976 to ASTM D3803-1989. The conditions employed by ASTM D3803-1989 were selected to approximate operating or accident conditions of a nuclear reactor which would severely reduce the performance of activated carbons. The ASTM D3803-1989 testing is more stringent than that required by the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, 1976. The testing temperature of ASTM D3803-1989 [is] lower than that of Regulatory Guide 1.52 and the relative humidity required by ASTM D3803-1989 is higher than that required by Regulatory Guide 1.52. This makes the ASTM D3803-1989 test being [sic] a more conservative test [than] that required by the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, 1976. Additionally, the proposed acceptance criteria of < 5.0% penetration of methyl iodide ensures that General Design Criterion 19 dose limits for control room operators are not exceeded. As such, the proposed license amendments do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Gordon E. Edison, Acting

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 amendments request: May 23, 1997

Description of amendments request: The proposed amendments to Technical Specification 3/4.4.5 for the Brunswick Steam Electric Plant, Units 1 and 2, reduce the short-term limit for Dose Equivalent I-131 activity in the reactor coolant from 4.0 microcuries/gram to 3.0 microcuries/gram. With coolant specific activity greater than 0.2 microcuries/gram Dose Equivalent I-131 but less than or equal to the short-term limit, operation of the affected unit may continue for up to 48 hours provided that operation under these conditions does not exceed 10 percent of the unit's total yearly operating time. With coolant specific activity greater than 0.2 microcuries/gram I-131 Dose Equivalent for more than 48 hours during one continuous time interval or greater than the short-term limit, the affected unit must be placed in Hot Shutdown within 12 hours. The purpose of the reduction of the short-term limit is to ensure control room operator dose following a Main Steam Line Break event is within the guidelines contained in 10 CFR Part 100 and the limits contained in Criterion 19 of Appendix A to 10 CFR Part 50.

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

The proposed amendments conservatively revise Action Statements a.1 and a.2 of Technical Specification 3/4.4.5 by reducing the maximum allowed reactor coolant specific activity from 4.0 to 3.0 [microcuries]/gram dose equivalent I-131. The purpose of the maximum allowable iodine specific activity is to ensure that the thyroid dose from a main steam line break (MSLB)is within the 10 CFR 100 dose guidelines and the General Design Criteria 19 dose limits for control room operators. The maximum allowable iodine specific activity is not associated with any initiating event and, therefore, cannot affect the probability of any accident. The proposed amendments result in a more conservative action limit and, therefore, do not increase the consequences of any accident.

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

The proposed amendments conservatively reduce the maximum allowable reactor coolant iodine specific activity. The activity limit is not associated with any initiating event and the system design is not affected. Therefore, the proposed amendments cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments revise Action Statements a.1 and a.2 of Technical Specification 3/4.4.5 by reducing the maximum allowed reactor coolant specific activity from 4.0 to 3.0 [microcuries]/gram dose equivalent I-131. As stated above, the purpose of the maximum allowable iodine specific activity is to ensure that the thyroid dose from a MSLB is within the 10 CFR 100 dose guidelines and the General Design Criteria 19 dose limits for control room operators. The reduction in the activity limit is a conservative change and, therefore, the proposed license amendments do not involve a 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 North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Gordon E. Edison, Acting

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

Date of amendment request: June 12, 1997

Description of amendment request: The amendment would make changes to the operations organization description.

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:

This change does not involve a significant hazards consideration for the following reasons:

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

The proposed amendment deals with changing position titles and clarification of the Harris Nuclear Plant (HNP) Operations management organization and responsibilities. The changes are considered to be admnistrative in nature and do not involve any modifications to any plant equipment or [affect] plant operation.

Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment deals with changing position titles and clarification of the HNP Operations management organization and responsibilities. The changes are considered to be administrative in nature and do not involve any modifications to any plant equipment or [affect] plant operation.

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

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

The proposed amendment does not reduce the margin of safety as defined in the Safety Analysis Report or the bases contained in the Technical Specifications. The requirement to have a licensed SRO [Senior Reactor Operator] management position responsible for plant operations is maintained within the proposed amendment.

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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Gordon E. Edison, Acting

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: May 27, 1997

Description of amendment request: The proposed amendments would revise Technical Specification Section 6, ''Administrative Controls,'' to incorporate revised organizational titles and would modify License Condition 2.C.(30)(a) to reflect that the Shift Technical Advisor function may be filled by someone other than a designated Senior Reactor Operator (SRO). In addition, the proposed amendments would change the submittal frequency of the Radiological Effluent Release Report from semiannually to annually. The proposed amendments will also make several administrative and editorial changes.

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

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

The proposed changes do not affect any accident initiators or precursors and do not change or alter the design assumptions for systems or components used to mitigate the consequences of an accident. The proposed changes do not affect the design or operation of any system, structure, or component in the plant. There are no changes to parameters governing plant operation, and, no new or different type of equipment will be installed.

The proposed changes provide clarification, consistency with station procedures, programs, the Code of Federal Regulations (10CFR), other Technical Specifications, and Improved Technical Specifications. These changes do not impact any accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. There is no relaxation of applicable administrative controls. Those administrative requirements which have no effect on safe operation of the plant are eliminated.

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

The proposed changes do not affect the design or operation of any plant system, structure, or component. There are no changes to parameters governing plant operation, and, no new or different type of equipment will be installed. The organizational and administrative changes proposed have no effect on the design or operation of any system, structure, or component in the plant. There are no changes to parameters governing plant operation; no new or different type of equipment will be installed.

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

The proposed changes do not affect the margin of safety for any Technical Specification. The initial conditions and methodologies used in the accident analyses remain unchanged; therefore, accident analyses results are not impacted. Plant safety parameters or setpoints are not affected. All responsibilities described in the Technical Specifications for administrative controls will continue to be performed by individuals possessing the requisite qualifications. Clarifications, relocations, and nomenclature changes neither result in a reduction of personnel responsibilities, nor do they cause a relaxation of programmatic controls. There are no resulting effects on plant safety parameters or setpoints.

Guidance has been provided in ''Final Procedures and Standards on No Significant Hazards Considerations,'' Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a purely administrative change to the Technical Specifications to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

The proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. The proposed change does not reduce the margin of safety as defined in the basis for any Technical Specification.

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

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

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

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: July 1, 1997

Description of amendment request: The proposed amendments would change the definition of Channel Calibration in section 1.4 of the Technical Specifications to require an inplace qualitative assessment of thermocouple and resistance temperature detectors which cannot be calibrated. The proposed amendments will also correct typographical and miscellaneous errors in TS Table 3.3.2-1, Table 3.3.6-1, and Bases section 3/4.3.1.

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

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

a. The change in the definition of a Channel Calibration is to make the wording more clear and to require an inplace qualitative assessment in place of the calibration of thermocouple and resistance temperature detector (RTD) sensors. The thermocouple and RTD sensors are not adjustable and are not subject to drift due to their design. The inplace qualitative assessments will assure proper functioning of the sensors, due to the nature of these sensors and the associated failure modes, and thus will verify that the sensors will be able to fulfill their intended function(s). Therefore the change to the definition will not change the probability or consequences of an accident previously evaluated.

b. Manual initiation of isolation actuation instrumentation trip systems for inboard and outboard valves is required to be operable per TS Table 3.3.2-1, Trip Functions B.1 and B.2, respectively. Trip Function B.2, outboard valves, lists valve group 7, TIP system isolation valves. Valve group 7 consists of an automatic inboard isolation valve for each TIP guide tube penetrating the primary containment (correctly listed under B.1), and a manual outboard isolation valve on each guide tube, that is an explosive squib valve. Each explosive squib valve is manually actuated with a keylock switch from the main control room per design. Each is a positive control backup upon failure of an inboard valve in the open position. The squib valves are not actuated from isolation actuation channel logic. This configuration meets the current design and licensing basis. Therefore, deletion of valve group 7 from TS Table 3.3.2-1 will not change the probability or consequences of an accident previously evaluated.

c. The proposed change to TS Table 3.3.6-1, Control Rod Withdrawal Block Instrumentation, deletes Note (e) from Trip Function 4.a, IRM detector-not-full-in rod block. This rod withdrawal block functions during Operational Condition 2, Startup, and 5, Refuel, to assure that IRMs are operable during control rod withdrawal in these plant Operational Conditions. The rod block is not bypassed when the IRMs are on range 1. Thus Note (e) does not apply to this trip function and is being deleted. Therefore, the correction of this error will not change the probability or consequences of an accident previously evaluated.

d. The change to TS Bases 3/4.3.1 to correct a typographical error referencing TS Table 3.3.1-2, Note ι, instead of Note ιι is an administrative change and thus will not change the probability or consequences of an accident.

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

The changes to the definition of Channel Calibration and correction of the other miscellaneous errors in the TS and TS Bases will not create the possibility of a new or different kind of accident, because the changes will not affect the design or operation of any structure, system, or component in the plant.

3) Involve a significant reduction in the margin of safety because:

a. The definition of Channel Calibration is being changed to be like the definition in NUREG 1434, Standard Technical Specifications General Electric Plants, BWR/6, Revision 1. The primary changes involve requiring only an inplace qualitative assessment of thermocouple and RTD sensors. These sensors are not adjustable and not susceptible to setpoint drift. Thus the appropriate check of the sensors is a qualitative assessment only. The inplace qualitative assessment assures operability of the sensors. Therefore there is no reduction in the margin of safety.

b. The remaining miscellaneous changes are corrections due to errors in the TS. The corrections will make the associated TS consistent with the design and licensing basis of LaSalle or correct typographical errors. Therefore, there is no 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 requested amendments involve no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: July 17, 1996, as supplemented by letters dated June 3, and July 7, 1997.

Description of amendment request: The proposed change request modifies Waterford Steam Electric Station, Unit 3, Technical Specifications (TSs) 3/4.7.1.3, ''CONDENSATE STORAGE POOL,'' by increasing the minimum Condensate Storage Pool (CSP) level from 82 percent to 91 percent in Modes 1, 2, and 3. The July 7, 1997, supplement proposes to expand the applicability of TS 3.7.1.3 to include Mode 4 operational requirements and maintains the 91 percent minimum CSP level previously requested for Modes 1, 2, and 3. The staff previously issued No Significant Hazard Considerations notice on March 26, 1997 (62 FR 14461).

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?Response: No.

Increasing the minimum required Condensate Storage Pool (CSP) level to 91 percent will insure that the minimum required 170,000 gallons of water is available to supply the Emergency Feedwater System and that 3,500 gallons of water is available for use by the Component Cooling Water Makeup System in Modes 1, 2, and 3. Maintaining a minimum required CSP level of 11 percent will insure that 3,500 gallons of water is available for use by the Component Cooling Water Makeup System in Mode 4. Maintaining the minimum required water volume will not increase the probability of any accident previously evaluated. Additionally, it will not affect the consequences of any accident. Maintaining a minimum required CSP level will ensure that the system remains within the bounds of the accident analysis. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Response: No.

Increasing the minimum water volume of the CSP from 82 percent to 91 percent in Modes 1, 2, and 3 does not create a possibility for a new or different kind of accident. Maintaining a minimum water volume of the CSP at 11 percent in Mode 4 does not create a possibility for a new or different kind of accident. The CSP will be operated in the same manner as previously evaluated. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Operation in accordance with this proposed change will ensure that the minimum contained water volume of the CSP will remain adequate under all conditions. This will improve the present 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, LA 70122

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

NRC Project Director: James W. Clifford, Acting Director

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

Date of amendment request: May 29, 1997

Description of amendment request: The proposed amendments will improve consistency throughout the Technical Specifications and their related Bases by removing outdated material, incorporating minor changes in text, making editorial corrections, and resolving other inconsistencies identified by the licensee's plant operations staff.

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

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

The proposed amendments consist of administrative changes to the Technical Specifications (TS) for St. Lucie Units 1 and 2. The amendments will implement minor changes in text to rectify reference, typographic, spelling, and/or consistency-in-format errors; update the TS Bases; and/or otherwise improve consistency within the TS for each unit. The proposed amendments do not involve changes to the configuration or method of operation of plantequipment that is used to mitigate the consequences of an accident, nor do the changes otherwise affect the initial conditions or conservatisms assumed in any of the plant accident analyses. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed administrative revisions will not change the physical plant or the modes of plant operation defined in the Facility License for each unit. 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 amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments are administrative in nature and do not change the basis for any technical specification that is related to the establishment of, or the preservation of, a nuclear safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: December 2, 1996

Description of amendment request: The proposed amendment would relocate the audit frequency requirements from the plant Technical Specifications to the Quality Assurance Plan. In addition, the maximum interval between certain types of audits will be extended. This change would make the TMI-2 technical specifications consistent with the Technical Specifications for Three Mile Island, Unit 1.

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:

10 CFR 50.92 provides the criteria which the Commission uses to perform a No Significant Hazards Consideration. 10 CFR 50.92 states that an amendment to a facility license involves No Significant Hazards if 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 proposed change to the technical specifications is administrative and does not involve any physical changes to the facility. No changes are made to operating limits or parameters, nor to any surveillance activities. Based on this, GPU Nuclear has concluded that the proposed change does not:

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

The proposed amendment is administrative and does not affect the function of any system or component. Therefore this change does not increase the probability of occurrence or the consequences of an accident previously evaluated.

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

The proposed change is administrative and no new failure modes or potential accident scenarios are created.

3. Involve a change in the margin of safety.

This change is administrative in nature and does not affect any safety settings, equipment, or operational parameters.

Based on the above analysis it is concluded that the proposed changes involve no significant safety hazards considerations as defined by 10 CFR 50.92.

The NRC staff has reviewed the analysis of the licensee 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: Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW, Washington, D.C. 20037 NRC Project Acting Director: Marvin M. Mendonca

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

Date of amendment request: July 16, 1997

Description of amendment request: The proposed amendment would revise Technical Specification Table 2.2-1 and 3/4.2.5 to allow the reactor coolant system total flow to be determined using cold leg elbow tap differential pressure measurements.

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:

Pursuant to 10[]CFR[]50.92 each application for amendment to an operating license must be reviewed to determine if the proposed change involves a Significant Hazards Consideration. The amendment, as defined below, describing the Technical Specification change associated with the change has been reviewed and determined to not involve Significant Hazards Considerations. The basis for this determination follows.

Proposed Change: The current Technical Specification Table 2.2-1 (page 2-4) ''Reactor Trip System Instrumentation Trip Setpoints,'' provides the Trip Setpoint and Allowable Value for the RCS [reactor coolant system] Flow-Low trip. The Allowable Value will be changed to reflect the increased uncertainty associated with the correlation of the elbow taps to a previous baseline calorimetric. In addition, Technical Specification 3.2.5 (page 3/4.2-11), ''Power Distribution Limits, DNB Parameters'', will be changed to allow the RCS total flow to be measured by the elbow tap [delta]p method. These changes will include the modification of surveillance requirement 4.2.5.3, which currently requires performance of a precision heat balance every 18 months, to allow use of the elbow tap [delta]p method for RCS flow measurement. Appropriate Technical Specification Bases sections will also be revised to reflect use of the elbow tap [delta]p method for flow measurement and to provide clarification. The revised Technical Specifications are in Appendix C.

Background: The 18-month total RCS flow surveillance is typically satisfied by a secondary power calorimetric-based RCS flow measurement. In recent cycles, South Texas Project has experienced apparent decreases in flow rates which have been attributed to variations in hot leg streaming effects. These effects directly impact the hot leg temperatures used in the precision calorimetric, resulting in the calculation of low RCS flow rates. The apparent flow reduction has become more pronounced in fuel cycles which have implemented aggressive low leakage loading patterns. Evidence that the flow reduction was apparent, but not actual, was provided by elbow tap measurements. The results of this evaluation, including a detailed description of the hot leg streaming phenomenon, are documented in Westinghouse report SAE/FSE-TGX/THX-0152, ''RCS Flow Verification Using Elbow Taps.''

South Texas Project intends to begin using an alternate method of measuring RCS flow using the elbow tap [delta]p measurements. For this alternate method, the RCS elbow tap measurements are correlated to precision calorimetric measurements performed during earlier cycles which decreased the effects of hot leg streaming.

The purpose of this evaluation is to assess the impact of using the elbow tap [delta]p measurements as an alternate method for performing the 18-month RCS flow surveillance on the licensing basis and demonstrate that it will not adversely affect the subsequent safe operation of the plant. This evaluation supports the conclusion that implementation of the elbow tap [delta]p measurement as an alternate method of determining RCS total flow rate does not represent a significant hazards consideration as defined in 10[]CFR[]50.92.

Evaluation: Use of the elbow tap [delta]p method to determine RCS total flow requires that the [delta]p measurements for the present cycle be correlated to the precision calorimetric flow measurement which was performed during the baseline cycle(s). A calculation has been performed to determine the uncertainty in the RCS total flow using this method. This calculation includes the uncertainty associated with the RCS flow baseline calorimetric measurement, as well as uncertainties associated with [delta]p transmitters and indication via QDPS [qualified display processing system] or the plant process computer. The uncertainty calculation performed for this method of flow measurement is consistent with the methodology recommended by the Nuclear Regulatory Commission (NUREG/CR-3659, PNL-4973, 2/85). The only significant difference is the assumption of correlation to a previously performed RCS flow calorimetric. However, this has been accounted for by the addition of instrument uncertainties previously considered to be zeroed out by the assumption of normalization to a calorimetric performed each cycle. Based on these calculations, the uncertainty on the RCS flow measurement using the elbow tap method is 2.6% flow which results in a minimum RCS total flow of 391,500 gpm and must be measured via indication with QDPS or the plant process computer at approximately 100% power.

The specific calculations performed were for Precision RCS Flow Calorimetrics for the specified baseline cycles, Indicated RCS Flow (either QDPS or the plant process computer), and the Reactor Coolant Flow - Low reactor trip. The calculations for Indicated RCS Flow and Reactor Coolant Flow - Low reactor trip reflect correlation of the elbow taps to baseline precision RCS Flow Calorimetrics. As discussed above, additional instrument uncertainties were included for this correlation.

The uncertainty associated with the RCS Flow - Low trip increased slightly. It was determined that due to the availability of margin in the uncertainty calculation, no change was necessary to either the Trip Setpoint (91.8% flow) or to the current Safety Analysis Limit (87% flow) to accommodate this increase. The Allowable Value is to be modified to allow for the increased instrument uncertainties associated with the [delta]p to flow correlation.

Since the flow uncertainty did not increase over the currently analyzed value, no additional evaluations of the reactor core safety limits must be performed. In addition, it was determined that the current Minimum Measured Flow (MMF) assumed in the safety analyses (389,200 gpm) bounds the required MMF calculated for the elbow tap method (391,500 gpm).

Based on these evaluations, the proposed change would not invalidate the conclusions presented in the UFSAR [Updated Final Safety Analysis Report].

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

Sufficient margin exists to account for all reasonable instrument uncertainties; therefore, no changes to installed equipment or hardware in the plant are required, thus the probability of an accident occurring remains unchanged.

The initial conditions for all accident scenarios modeled are the same and the conditions at the time of trip, as modeled in the various safety analyses, are the same. Therefore, the consequences of an accident will be the same as those previously analyzed.

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

The proposed change revises the method for RCS flow measurement, and therefore does not introduce any new accident indicators or failure mechanisms.

No new accident scenarios have been identified. Operation of the plant will be consistent with that previously modeled, i.e., the time of reactor trip in the various safety analyses is the same, thus plant response will be the same and will not introduce any different accident scenarios that have not been evaluated.

3. Does the proposed modification involve a significant reduction in a margin of safety[?]

There are no changes to the Safety Analysis assumptions. Therefore, the margin of safety will remain the same.

The proposed change does not impact the results from any accidents analyzed in the safety analysis.

Conclusion: Based on the preceding information, it has been determined that this proposed change to allow an alternate RCS total flow measurement based on elbow tap [delta]p measurements does not involve a Significant Hazards Consideration as defined by 10 CFR 50.92(c).

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

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

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

NRC Project Director: James W. Clifford, Acting

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 1, Oswego County, New York

Date of amendment request: July 2, 1997

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.2.3 regarding reactor coolant chemistry in accordance with a report by Electrical Power Research Institute, Inc. (EPRI) TR-103515-R1, ''BWR Water Chemistry Guidelines, 1996 Revision,'' also known as Boiling Water Reactor Vessel and Internals Project (BWRVIP)-29. Specifically, the amendment would define new conductivity limits in TS 3.2.3a (when reactor coolant is 200 degrees F or more and reactor thermal power is no more that 10%), and in TS 3.2.3b (when reactor thermal power exceeds 10%). The new conductivity limits would be 1 micro-mho/cm, which is less than the existing limits of 2 micro-mho/cm and 5 micro-mho/cm. The chloride ion limit in TS 3.2.3a, 0.1 ppm, would remain at this value but would be designated as 100 ppb. The chloride ion limit in TS 3.2.3b would be changed from 0.2 ppm to 20 ppb. Sulfate ion limits would be added to TS 3.2.3a and TS 3.2.3b at 100 ppb and 20 ppb, respectively. In TS 3.2.3c, the maximum conductivity limit would be changed from 10 micro-mho/cm to 5 micro mho/cm when reactor coolant temperature is 200 degrees F or more; the maximum chloride ion concentration limit would be changed from 0.5 ppm to 100 ppb (when reactor thermal power exceeds 10%) and 200 ppb (when reactor coolant temperature is 200 degrees F or more and reactor thermal power is no more than 10%); and the maximum sulfate ion concentration of 100 ppb (when reactor thermal power exceeds 10%) and 200 ppb (when reactor coolant temperature is 200 degrees F or more and reactor thermal power is no more than 10%) would be added. The requirement to place the reactor in the cold shutdown condition as currently specified in TS 3.2.3d (when TSs 3.2.2a, b, and c are not met) and TS 3.2.3e (when the continuous conductivity monitor is inoperable for more than 7 days) would be changed to require that the reactor coolant temperature be reduced to below 200 degrees F. TS 4.2.3 would be revised to add that the samples taken and analyzed for conductivity and chloride ion content are also to be analyzed for sulfate ion content. TS Bases 3/4.2.3 would also be changed to reflect that the purpose of TS 3/4.2.3 is to limit crack growth rates to values consistent with Unit 1 core shroud analyses in accordance with an NRC letter dated May 8, 1997.

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 operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes to the conductivity and chloride ion action levels and the addition of sulfate ion levels as an action level in reactor water chemistry are being made to make the TS and its Bases consistent with the values used in the core shroud vertical weld cracking evaluations. These new values reflect the BWR water chemistry guidelines, 1996 revision (EPRI TR-103515-R1, BWRVIP-29) and are equal to or more restrictive than the present TS values. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. None of the precursors of previously evaluated accidents are affected and therefore, the probability of an accident previously evaluated is not increased. These changes to the coolant chemistry TS are more restrictive limits and no new failure modes are introduced. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes to the conductivity and chloride ion action levels and the addition of sulfate ion levels as an action level in reactor water chemistry are being made to make the TS and its Bases consistent with the values use in the core shroud vertical weld cracking evaluations. The new values reflect the BWR water chemistry guidelines, 1996 revision (EPRI TR-103515-R1, BWRVIP-29) and are equal to or more restrictive than the present TS values. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. The change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, this change does not create the possibility of a new or different kind of accident [from any accident] previously evaluated.

3. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The changes to the conductivity and chloride ion action levels and the addition of sulfate ion levels as an action level in reactor water chemistry are being made to make the TS and its Bases consistent with the values used in the core shroud vertical weld cracking evaluations. These new values reflect the BWR water chemistry guidelines, 1996 revision (EPRI TR-103515-R1, BWRVIP-29) and are equal to or more restrictive than the present TS values. No physical modification of the plant is involved and no changes to the methods in which plant systems are operated are required. This change does not adversely affect any physical barrier to the release of radiation to plant personnel or the public. Therefore, the change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126

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

NRC Project Director: Alex Dromerick, Acting

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

Date of amendment request: June 19, 1997

Description of amendment request: Technical Specification Table 2.2-1 Notes 1 and 3 define the values for the constants used in the Overtemperature Delta-T and Overpower Delta-T reactor trip system instrumentation setpoint calculators. The proposed amendment would make changes to the notes as well as the associated Bases section.

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:

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed revision does not involve an SHC because the revision would not:

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

The proposed changes to Technical Specification Table 2.2-1 Notes 1 and 3 for the addition of the inequalities ensure that the constants used for [Overtemperature Delta-T] and [Overpower Delta-T] will be set conservatively with respect to the assumptions in the accident analysis. The effect on the turbine

runback function has been evaluated with respect to the Loss of External Electrical Load And/Or Turbine Trip analysis and it has been determined that this change does not increase the probability of this transient. The change was also reviewed to determine if it produced an increase in the probability of an unnecessary or spurious reactor trip and it was determined that it did not. This change does not increase the probability of any previously evaluated accident.

The consequences of previously evaluated accidents, including Uncontrolled Rod Cluster Assembly Bank Withdrawal At Power, Rod Cluster Control Assembly Misalignment, Uncontrolled Boron Dilution, Loss of External Electrical Load And/Or Turbine Trip, Excessive Heat Removal Due To Feedwater System Malfunctions, Excessive Load Increase Incident, Accidental Depressurization Of The Reactor Coolant System, Accidental Depressurization Of The Main Steam System, Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuate ECCS [emergency core cooling system], or Major Secondary System Pipe Ruptures have not changed.

The administrative changes have no impact on the design or operation of Millstone Unit 3.

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

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

The proposed changes to Technical Specification Table 2.2-1 Notes 1 and 3 do not alter the design, construction, operation, maintenance or method of testing of equipment. The proposed changes alter the Technical Specification description of [an] [Overtemperature Delta-T] and [Overpower Delta-T] setpoint functions and requires only slight changes to the actual setpoints in the field. The [Overtemperature Delta-T] and [Overpower Delta-T] functions serve to mitigate the effects of accidents by opening the Reactor Trip breakers or reduce power by ''running back'' turbine electrical load. The change does not create any new interfaces to plant control or protection systems and therefore, no new mechanism for accident initiation has been introduced. The proposed change does not introduce the possibility of an accident of a different type than previously evaluated.

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

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

The proposed changes to Technical Specification Table 2.2-1 Notes 1 and 3 do not affect the integrity of any physical fission protective boundaries, increase the delays in actuation of safety systems beyond that assumed in the safety analysis or reduce the margin of safety of any system. These changes ensure that actuation of Overtemperature [Delta-T] and Overpower [Delta-T] reactor trips will occur conservatively with respect to the assumptions of the accident analysis.

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

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: June 19, 1997

Description of amendment request: Technical Specification 3/4.7.1.3 requires sufficient water to be available for the auxiliary feedwater (AFW) system to maintain the reactor coolant system at hot standby for 10 hours before cooling down to hot shutdown in the next 6 hours. The proposed amendment would increase the required volume of water when the condensate storage tank is used, make editorial changes, and expand the description in the appropriate Bases section.

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:

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed revision does not involve an SHC because the revision would not:

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

The proposed change to Technical Specification Surveillance 4.7.1.3.2 will account for the unusable Condensate Storage Tank (CST) inventory by increasing the required combined CST and Demineralized Water Storage Tank (DWST) inventory to 384,000 gallons. The increased required water volume is consistent with the design of the CST and will provide assurance that sufficient water is available to maintain the reactor coolant system at Hot Standby for 10 hours before cooling down to Hot Shutdown in the next 6 hours.

The proposed changes to reword Technical Specification 3/4.7.1.3, expand the description in Bases Section B3/4.7.1.3 and modify the description in Bases Section B3/4.7.1.2 are to update and clarify the requirements.

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

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

The proposed changes to Technical Specification 3/4.7.1.3 do not change the use of DWST or CST during normal or accident evaluations.

The proposed changes to reword Technical Specification 3/4.7.1.3, Bases Section B3/4.7.1.3 and Bases Section B3/4.7.1.2 are to update and clarify the requirements.

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

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

The proposed changes to Technical Specification Surveillance 4.7.1.3.2 will increase the required inventory for the combined CST and DWST to account for an additional 50,000 gallons of unusable inventory due to the CST discharge line location, other physical characteristics, and measurement uncertainty. The proposed change to the surveillance requirement will increase the required volume of the combined CST and DWST inventory to 384,000 gallons. The proposed change ensures that sufficient water is available to maintain the Reactor Coolant System at Hot Standby conditions for 10 hours with steam discharge to the atmosphere, concurrent with a total loss-of-offsite power, and with an additional 6-hour cool down period to reduce reactor coolant temperature to 350 [degrees] F.

The proposed changes to Technical Specification 3/4.7.1.3 and Bases Section 3/4.7.1.3 are to clarify the requirements. The proposed changes to the Bases Section 3/4.7.1.2 update and expands the description of the design bases accidents for which AFW System is credited for accident mitigation. This additional information is consistent with the current AFW System design bases.

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

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: June 30, 1997

Description of amendment request: Technical Specification Surveillance Requirements 4.7.1.5.1 and 4.7.1.5.2 require the periodic testing of the main steam isolation valves (MSIVs) to demonstrate operability. The proposed amendment would (1) clarify when the MSIVs are partial stroked or full closure tested, (2) add a note to the Mode 4 applicability of Technical Specification 3.7.1.5 to require that the MSIVs be closed and deactivated at less than 320 degrees F, (3) make editorial changes, and (4) make changes to the associated Bases sections.

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:

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

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

The proposed changes to Technical Specifications Surveillances 4.7.1.5.1 and 4.7.1.5.2 are to clarify the testing of the MSIVs by rewording and separating the requirements into three surveillances.

Currently, Technical Specifications Surveillance 4.7.1.5.1 requires ''verifying full closure within 10 seconds ... in MODES 1, 2, and 3 when tested pursuant to Specification 4.0.5.'' The current surveillance requirement to full stroke test the MSIVs is not performed during power operation as the Millstone Unit 3 Inservice Pump and Valve Test Program pursuant to Specification 4.0.5, has received relief from the quarterly full stroke surveillance testing requirement. The basis for the relief is that full stroking the MSIVs to the closed position during power operation would result in an unbalanced steam flow condition producing an abnormal power distribution in the reactor core, possibly causing a reactor trip. The MSIVs are equipped with provisions for inservice testing by partial stroking. The partial stroking is accomplished by opening a solenoid valve to admit steam pressure into the lower piston chamber. After a time delay the solenoid valve for the upper piston chamber opens. After 10 percent travel the position indicating device vents both piston chambers and the valve fully opens to the back seat due to pressure acting on the valve plug. The accepted alternate testing method is to partially stroke test the MSIVs during power operation and full stroke test the valves during shutdowns.

The proposed changes to Technical Specifications Surveillance 4.7.1.5.2 will identify a Mode 3 requirement to perform a 10 second full closure test of the MSIVs in Mode 3 or 4. Surveillance 4.7.1.5.3 will identify a Mode 4 requirement to perform a 120 second full closure test of the MSIVs in Mode 4 when the RCS [reactor coolant system] temperature is greater than or equal to 320 degrees F. The 320 degrees F restriction on testing the valves is consistent with recommendations from the valve manufacturer. Additionally, a footnote is added to the LCO [limiting condition for operation] and the surveillance to identify that the MSIVs are required to be closed and deactivated when the RCS temperature is less than 320 degrees F.

The proposed changes are consistent with equipment design and the surveillance testing of the MSIVs provides the necessary assurance that the valves will function consistent with accident analyses.

The other proposed changes to reword the Applicability and Action statements of Technical Specification 3.7.1.5 and Bases Section B3/4.7.1.5 are considered administrative changes.

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

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

The proposed changes to the surveillance testing of the MSIVs does not change the operation of the valves as assumed for accident analyses. The MSIVs are currently equipped with provisions for partial stroking.

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

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

The proposed changes to Technical Specifications Surveillances 4.7.1.5.1 and 4.7.1.5.2 are to clarify the testing of the MSIVs by rewording and separating the requirements into three surveillances. Surveillance 4.7.1.5.1 will identify a Mode 1 and 2 requirement to partial stroke test the MSIVs in Mode 1 and 2 unless a successful 10 second full stroke test was performed during the surveillance period. Surveillance 4.7.1.5.2 will identify a Mode 3 requirement to perform a 10 second full closure test of the MSIVs in Mode 3 or 4. Surveillance 4.7.1.5.3 will identify a Mode 4 requirement to perform a 120 second full closure test of the MSIVs in Mode 4 when the RCS temperature is greater than or equal to 320 degrees F. The 320 degrees F restriction on testing the valves is consistent with recommendations from the valve manufacturer. Additionally, a footnote is added to the LCO and the surveillance to identify that the MSIVs are required to be closed and deactivated when the RCS temperature is less than 320 degrees F. The footnote will eliminate the potential to declare the MSIVs operable in the upper range of Mode 4 and then allow the MSIVs to remain open during a cooldown into the lower range of Mode 4 where they may not be able to meet their required stroke time. The full closure test times are consistent with the current MSIV surveillances and the partial stroke testing is consistent with the Millstone Unit 3 Inservice Pump and Valve Test Program.

The other proposed changes to reword the Applicability and Action statements of Technical Specification 3.7.1.5 and Bases Section B3/4.7.1.5 are considered administrative changes.

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

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: June 30, 1997

Description of amendment request: Technical Specifications 4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3 require the testing of the containment to verify leakage limits at a specified test pressure. The proposed amendment would (1) modify the list of valves that can be opened in Modes 1 through 4, (2) add a footnote on procedure controls, (3) remove a footnote on Type A testing, and (4) make editorial changes to the Technical Specifications and associated Bases sections.

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:

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

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

The proposed changes to Technical Specification Surveillance 4.6.1.1.a include the adding ''or procedure control***'' and adding footnote ''***''. The changes are requested since the Residual Heat Removal System (RHR) valves, 3RHS*MV8701A/B and 3RHS*MV8702A/B, are opened during cooldown and heatup in Mode 4. Allowing these containment isolation valves to be opened is consistent with Technical Specification 3.4.1.3, Reactor Coolant System - Hot Shutdown, which allows the RHR system to be used in Mode 4. The proposed changes to open the RHR system containment isolation valves, under procedure control in Mode 4, do not change the way the RHR system is operated or change the operator's response to an accident in Mode 4.

The proposed changes to Technical Specification Surveillance 4.6.1.1.a Footnote **, include the modification of the valves listed in the footnote. Valves 3FPW-V661, 3FPW-V666, 3SAS-V875, 3SAS-V50, 3CCP-V886, 3CCP-V887 and 3CVS-V13 are being deleted and are local manual containment isolation valves. Deleting these valves from the list of valves that are allowed to be opened under administrative control does not modify plant response to or mitigation strategy for any accident. The valves being added, 3MSS*V885, 3MSS*V886, and 3MSS*V887, are in the steam lines to the steam-driven auxiliary feedwater pump. These valves are opened to warm the steam lines prior to testing the steam-driven auxiliary feedwater pump. These valves were recently reclassified as containment isolation valves, which resulted in the need to add them to the list of valves allowed to be opened under administrative control. The administrative controls include the appropriate considerations that when required, containment integrity will be established consistent with the assumptions in the design basis analyses.

The proposed change to Technical Specification Surveillance 4.6.1.2.a will delete footnote ‘‘*’’ which referred to an exemption granted by the NRC to permit the Type A test to be delayed until RFO6 [refueling outage 6]. However, the current extended shutdown has significantly delayed RFO6 and NNECO intends to perform the Type A test during this midcycle shutdown. The deletion of the footnote does not alter the operation of any system or the containment or containment airlocks, as assumed for accident analyses.

Additionally, Technical Specifications 4.6.1.1, 3/4.6.1.2 and 3/4.6.1.3 and Bases Sections 3/4.6.1.1, 3/4.6.1.2 and 3/4.6.1.3 are reworded to provide clarity and consistency. These proposed changes do not alter the operation of any system or the containment or containment airlocks during accident analyses. Therefore, the proposed revision does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed changes to Technical Specifications 4.6.1.1, 3/4.6.1.2 and 3/4.6.1.3 and Bases Sections 3/4.6.1.1, 3/4.6.1.2 and 3/4.6.1.3 do not alter the operation of any system or the containment or containment airlocks, during normal operation or as assumed in accident analyses.

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

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

The proposed changes to Technical Specifications 4.6.1.1, 3/4.6.1.2 and 3/4.6.1.3 and Bases Sections 3/4.6.1.1, 3/4.6.1.2 and 3/4.6.1.3 do not alter the design, maintenance or function of any system or the containment or the containment airlocks. Additionally, the proposed changes do not alter the testing of any system or the containment or containment airlocks, or alter any assumption used in the accident analyses.

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

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Deputy Director: Phillip F. McKee

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

Date of amendment requests: May 14, 1997

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 by revising Technical Specification (TS) 6.9.1.8.b.5 to replace reference WCAP-10266-P-A with WCAP-12945-P for best estimate loss-of coolant accident (LOCA) analysis. The amendment would also revise TS Bases 3/4.2.2 and 3/4.2.3 to change the emergency core cooling system (ECCS) acceptance criteria limit to state that there is a high level of probability that the ECCS acceptance criteria limits are not exceeded. This is consistent with the best estimate LOCA methodology.

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

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

The proposed change to use of the Best Estimate Loss of Coolant Accident (LOCA) analysis methodology does not involve physical alteration of any plant equipment or change in operating practice at Diablo Canyon Power Plant (DCPP). Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased.

The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, paragraph b, and it meets the five criteria listed in Section D. of this evaluation. No other accident is potentially affected by this change.

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

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

The proposed change would not result in any physical alteration to any plant system, and there would not be a change in the method by which any safety related system performs its function. The parameters assumed in the analysis are within the design limits of existing plant equipment.

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

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

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the DCPP Units 1 and 2 reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been analyzed to provide assurance that the most severe postulated LOCAs were calculated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, paragraph b, are met.

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

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment requests: May 15, 1997

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2 to revise the surveillance frequencies from at least once every 18 months to at least once per refueling interval (nominally 24 months) including (1) reactor coolant system total flow rate, (2) instrumentation for radiation monitoring, (3) instrumentation and controls for remote shutdown, (4) instrumentation for accident monitoring, and (5) several miscellaneous TS.

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

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

The proposed TS surveillance interval increases do not alter the intent or method by which the inspections, tests, or verifications are conducted, do not alter the way any structure, system, or component functions, and do not change the manner in which the plant is operated. The surveillance, maintenance, and operating histories indicate that the equipment will continue to perform satisfactorily with longer surveillance intervals. Few surveillance and maintenance problems were identified. No problems have recurred, or are expected to recur, following identification of root causes and implementation of corrective actions.

There was one time-related degradation mechanism identified that could significantly degrade the performance of the evaluated equipment during normal plant operation. Accumulation of corrosion products and debris in the containment fan cooler unit (CFCU) monitoring system drain lines could affect the use of the CFCU drains as a backup to the containment gaseous monitor for RCS leak detection. Primarily because CFCU drain line cleaning has been instituted to reduce deposit buildup, and also because the CFCU monitoring systems are used as backup and they are redundant by a factor of five, it was evaluated that this time-related mechanism will not significantly degrade the leak detection performance of the CFCUs.

All other potential time-related degradation mechanisms have insignificant effects in the period of interest (24 months plus 25 percent allowance, or a maximum of 30 months). Instrument drift and uncertainty analyses show that, while slight increases in instrument drift can occur over a longer period, such increases are minimal and remain within specified instrument accuracy and calibration allowable values. In cases (pressurizer water level and RVLIS) where greater than expected instrument drift has been found, design and procedural changes have been implemented to improve the calibration process and instrument performance. Based on the past performance of the equipment, the probability or consequences of accidents previously evaluated would not be significantly affected by the proposed surveillance interval increases.

The changes to commitments related to Bulletin 90-01 are supported by the conclusions above, and otherwise do not alter the intent or method by which the associated functions are tested, do not alter the way any structure, system, or component functions, and do not change the manner in which the plant is operated.

The administrative changes to the Bases sections and to remove a duplicate line do not alter the frequency, intent, or method by which the associated functions are tested, do not alter the way any structure, system, or component functions, and do not change the manner in which the plant is operated.

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

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

The surveillance and maintenance histories indicate that the equipment will continue to effectively perform its design function over the longer operating cycles. Additionally, the increased surveillance intervals do not result in any physical modifications, affect safety function performance or the manner in which the plant is operated, or alter the intent or method by which surveillance tests are performed. No problems have reoccurred following identification of root causes and implementation of corrective actions. Almost all identified potential time-related degradations, including instrument drift, have insignificant effects in the period of interest.

The deposit buildup in the CFCU drain lines is time-related. This was evaluated to not to be significant to the leak detection function because the CFCUs have a redundancy factor of five (any one of the five CFCUs can be used for the leak detection function) and because the CFCU drain lines will be cleaned each refueling outage. The proposed surveillance interval increases would not affect the type or possibility of accidents.

The changes to commitments related to Bulletin 90-01 are supported by the conclusions above, and otherwise do not result in any physical modifications, affect safety function performance or the manner in which the plant is operated, or alter the intent or method by which surveillance tests are performed.

The administrative change to the Bases sections and to remove a duplicate line do not result in any physical modifications, affect safety function performance, or alter the frequency, intent, or method by which surveillance tests are performed.

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

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

Evaluation of historical surveillance and maintenance data indicates that there have been few problems experienced with the evaluated equipment. There are no indications that potential problems would be cycle-length dependent, with the exception of the CFCU leak detection function, or that potential degradation would be significant for the period of interest and, therefore, increasing the surveillance interval will have negligible impact on safety. The accumulation of corrosion products and debris in the CFCU drain lines is cycle-length dependent, but has been evaluated to have insignificant effect on its leak detection function. There is no safety analysis impact since these changes will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria. Safety margins are not significantly impacted by surveillance intervals or by the slight increases in instrument drift that may occur during the extended interval.

The changes to commitments related to Bulletin 90-01 are supported by the conclusions above, and otherwise will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria.

The administrative change to the Bases sections and to remove a duplicate line will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria.

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

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

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

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

NRC Project Director: William H. Bateman

Pacific Gas and Electric Company, Docket No. 50-133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of amendment request: December 9, 1996, as supplemented on June 12, 1997.

Description of amendment request: The proposed amendment would revise the Humboldt Bay Power Plant (HBPP), Unit 3 Technical Specifications (TSs) to incorporate the requirements of 10 CFR Part 50, Appendix I, into the Radiological Effluent Technical Specifications (RETS) and to relocate the controls and limitations on RETS and radiological monitoring from the technical specifications to the Offsite Dose Calculation Manual (ODCM) and the Process Control Program (PCP). Additional minor administrative changes are proposed to make the TSs on High Radiation Areas consistent with the revised requirements in the new 10 CFR Part 20.

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?

Operation of the facility in accordance with the proposed amendment would not involve any increase in the probability or consequences of an accident previously evaluated. This change places new requirements in the Administrative Controls section of the Technical Specifications to establish programs for the control of radiological effluents and the conduct of radiological environmental monitoring in the ODCM. The new Administrative Control requirements for radiological effluents to be placed in the ODCM incorporate 10 CFR 50, Appendix I, limitations on dose to individual members of the public that are much more restrictive than the current Technical Specification limitations. The proposed changes do not involve modifications to existing plant equipment, the addition of new equipment, or operation of the plant in a different manner than previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability of 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?

Operation on the facility in accordance with the proposed amendment will not create any new or different kind of accident from any accident previously evaluated. As stated above, new programmatic controls on radiological effluents and radiological environmental monitoring are established in the Administrative Controls section of the Technical Specifications. Additionally, this change is administrative in nature; procedural details for radiological effluents and radiological environmental monitoring are being relocated to the ODCM and PCP consistent with the guidance provided [by the NRC] in Generic Letter 89-01. The proposed changes do not involve alterations to plant operating philosophy or methods, or in changes to installed plant systems, structures, or components.

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

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

Operation of the facility in accordance with the proposed amendment would not involve any reduction in the margin of safety. These changes do not involve a significant reduction in the margin of safety. These changes do not involve a significant reduction in the margin of safety. The changes will provide control over radiological effluent releases, solid waste management, and radiological environmental monitoring activities. Also, these changes will increase the margin of safety for members of the public by imposing additional controls to ensure that dose to members of the public resulting from radioactive effluent releases will be maintained ALARA.

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

The NRC staff has reviewed the analysis of the licensee 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: Humboldt County Library, 636 F Street, Eureka, California 95501

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

NRC Project Director: Seymour H. Weiss

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia

Date of amendment request: May 9, 1997

Description of amendment request: The proposed amendment would revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) in Technical Specification (TS) 2.1.1.2 to reflect results of a cycle-specific calculation performed for Unit 1 Operating Cycle 18 (expected to commence November 1997).

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

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

The derivation of the revised SLMCPR for Plant Hatch Unit 1 Cycle 18 for incorporation into the TS, and its use to determine cycle-specific thermal limits, have been performed using NRC approved methods. Additionally, interim implementing procedures that incorporate cycle-specific parameters have been used which result in a more restrictive value for SLMCPR. These calculations do not change the method of operating the plantand have no effect on the probability of an accident initiating event or transient.

The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR preserves the existing margin to transition boiling and the probability of fuel damage is not increased. Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes result only from a revised method of analysis for the Unit 1 Cycle 18 core reload. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined in the TS bases will remain the same. The new SLMCPR is calculated using NRC approved methods which are in accordance with the current fuel design and licensing criteria. Additionally, interim implementing procedures, which incorporate cycle-specific parameters, have been used. The SLMCPR remains high enough to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity.

Therefore, the proposed TS changes do not involve a 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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Herbert N. Berkow

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: May 9, 1997

Description of amendment request: The proposed amendments would revise the operability requirements for the Rod Block Monitor system of Technical Specification (TS) Table 3.3.2.1-1. The amendments would also delete the requirements of TS Section 5.6.5 to report Rod Block Monitor operability requirements in the cycle-specific Core Operating Limits Report.

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:

Southern Nuclear Operating Company has evaluated the proposed changes to the Plant Hatch Units 1 and 2 Technical Specifications in accordance with the criteria specified in 10 CFR 50.92 and has determined that they do not involve a significant hazards consideration because:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated since they are more restrictive than the existing requirements for operation of the plant. These changes provide assurance that the Rod Block Monitor system will remain operable when necessary to prevent or mitigate the consequences of an anticipated operational occurrence that could threaten the integrity of the fuel cladding integrity. Since changes in RBM [Rod Block Monitor] operability requirements do not involve any physical or functional modifications in any plant system, structure or component, there will be no increase in the probability or consequences of any accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated because they do not involve any changes in the plant configuration or in the operation of any system, structure or component.

3. The proposed changes do not reduce a margin of safety in the plant because they impose more restrictive operability requirements on the Rod Block Monitor system than those imposed by the existing specifications. The changes are more restrictive in that they delete the conditions under which the RBM is allowed to be bypassed at core thermal power equal to or greater than 29% of rated power. These more restrictive requirements ensure the RBM will not only prevent fuel rods from under going transition boiling, they also prevent fuel rods from exceeding 1% plastic strain (thereby avoiding fuel cladding damage) during an RWE [rod withdrawal error] event.

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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Herbert N. Berkow

Tennessee Valley Authority, Docket Nos. 50-260 and 50-296, Browns Ferry Nuclear Plant, Units 2 and 3, Limestone County, Alabama

Date of amendment request: June 19, 1997 (TS 391T)

Description of amendment request: The proposed amendment extends the allowed outage time for emergency diesel generators from 7 to 14 days on a one-time basis. This extension should permit completion of extensive recommended maintenance within a single outage interval.

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

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

The EDGs [emergency diesel generators] are designed as backup AC [alternating current] power sources in the event of loss of off-site power. The proposed AOT [allowed outage time] does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in the manner in which the EDGs provide plant protection or which create new modes of plant operation. Also, the TS [technical specification] change will improve the overall EDG availability by allowing the consolidation of planned maintenance outages and, hence, reducing the time period that each EDG will be in an outage. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce any new modes of plant operation or make physical changes to plant systems. Therefore, the proposed one-time extension of the allowable AOT for EDGs does not create the possibility of a new or different accident.

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

BFN's [Browns Ferry Nuclear Plant's] emergency AC system is designed with sufficient redundancy such that an EDG may be removed from service for maintenance or testing. The remaining EDGs are capable of carrying sufficient electrical loads to satisfy the UFSAR [updated final safety analysis report] requirements for accident mitigation or unit safe shutdown.

Since the 12-year EDG PM [preventive maintenance] work activity and vendor recommended PMs are required tasks which must be performed, the proposed TS would reduce EDG unavailability since multiple outages with resultant longer EDG outage times would not be necessary to accomplish the planned maintenance activities.

The proposed change does not impact the redundancy or availability requirements of off-site power supplies or change the ability of the plant to cope with station blackout events. The TS change improves overall EDG availability. For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: June 24, 1997

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.3.2.1, ''Safety Features Actuation System Instrumentation,'' TS Section 3/4.6.1.7, ''Containment Ventilation System,'' TS Section 3/4.6.3.1, ''Containment Isolation Valves,'' and TS Section 3/4.9.4, ''Refueling Operations - Containment Penetrations,'' and the associated TS Bases. Valve position requirements would be added, and certain containment radiation monitor requirements, valve isolation verification requirements, and containment radiation monitor optional uses would be deleted.

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:

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation ofthe Davis-Besse Nuclear Power

Station (DBNPS), Unit No. 1, in accordance with this change would:

1a Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are affected by the proposed changes.

The proposed changes to the Technical Specifications and their Bases ensure that during Modes 1 through 4 the Containment (CTMT) purge and exhaust isolation valves are closed with control power removed. Having these valves closed will not increase the probability of an accident because these valves are not accident initiators. They are used to mitigate the consequences of an accident. The proposed changes require these valves to be maintained in a closed position as required by design basis accident analysis.

The removal of the Safety Features Actuation System (SFAS) Radiation Monitors (RE's) and their associated SFAS Level 1 actuations does not affect any accident initiator, condition, or assumption.

During Modes 1 and 2 and partially in Mode 3, for design basis accidents which require CTMT isolation, the high/high-high CTMT pressure or low/low-low Reactor Coolant System (RCS) signals provide CTMT isolation and isolation and actuation of those components presently actuated by an SFAS Level 1 High Radiation signal. During Mode 3, when the RCS pressure is below 1800 psig, the low RCS pressure trip may be manually bypassed, and when the RCS pressure is below 600 psig, the low-low pressure trip may be manually bypassed. During the short period of time that these bypasses are activated in Mode 3, CTMT isolation is only automatically initiated by the CTMT high/high-high pressure trips. Manual SFAS actuation is also available, including Modes 1 through 4. Removing the SFAS RE's does not affect the operation of the SFAS Levels 2-4 actuation since these are based only on containment pressure and RCS pressure. Therefore, the assumption of CTMT isolation following design basis accidents is maintained.

The SFAS is not required in Mode 5. During Mode 6, the SFAS RE's and their associated SFAS Level 1 actuation are not credited during a fuel handling accident inside CTMT. The analysis for a fuel handling accident inside CTMT assumes that there is no isolation of CTMT. The probability of a fuel handling accident is not affected by these changes.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not change the source term, CTMT isolation, or allowable releases.

The proposed changes to the Technical Specifications and their Bases ensure that during Modes 1 through 4, the CTMT purge and exhaust isolation valves are closed with control power removed.

Having these valves closed and their control power removed ensures that the valves are in and will remain in, the proper position for CTMT isolation during and following design basis accidents. Also, during Modes 1 and 2 and partially in Mode 3, SFAS actuation on high/high-high CTMT pressure or low/low-low RCS pressure provides for diverse CTMT isolation. As noted above, during Mode 3, when the RCS pressure is below 1800 psig, the low RCS pressure trip may be manually bypassed, and when the RCS pressure is below 600 psig, the low-low pressure trip may be manually bypassed. During the short period of time that these bypasses are activated in Mode 3, CTMT isolation is only automatically initiated by the CTMT high/high-high pressure trips. In addition, manual SFAS actuation is also available, including during Modes 1 through 4. Therefore, removal of the SFAS RE's and their actuation signal does not prevent CTMT isolation.

The SFAS RE's and automatic isolation of the CTMT purge and exhaust isolation valves during a fuel handling accident is not required because the CTMT purge and exhaust isolation system, including the associated noble gas monitor, with operator action, can provide the necessary actions to mitigate a fuel handling accident inside CTMT, assuming the purge and exhaust valves are open. Therefore, removing the SFAS RE's and their actuation signal will not increase the consequences of an accident because CTMT closure is ensured. Further, it is noted that CTMT isolation is not assumed in the accident analysis for the fuel handling accident.

The Containment Radiation-High trip feature is not credited for any DBNPS Updated Safety Analysis Report (USAR) accident analysis, therefore the proposed removal of this feature will not impact radiological consequences of such accidents.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes.

As stated above, the CTMT purge and exhaust isolation valves, the SFAS RE's, and SFAS actuation are not accident initiators. Maintaining the CTMT purge and exhaust isolation valves closed and control power removed ensures that the design basis assumption of CTMT isolation is maintained. Also, since SFAS Levels 2-4 actuation, as applicable, on high/high-high CTMT pressure or low/low-low RCS pressure or by manual actuation provides the required diversity of sensing parameters and isolation of CTMT, the SFAS RE's and their associated automatic isolation of the CTMT purge and exhaust isolation valves is not required during Modes 1 through 4. Therefore, no new or different kind of accident will be introduced.

3. Not involve a significant reduction in a margin of safety because the proposed changes maintain a redundant and diverse CTMT isolation capability following design basis accidents. Under TS 3/4.3.2, diversity in achieving CTMT isolation by means of a high/high-high CTMT pressure or low/low-low RCS pressure SFAS actuation will be maintained during Modes 1 through 3 (except during brief periods of bypass in Mode 3), and the redundancy of the SFAS sensor instrumentation channels and actuation channels themselves will be maintained. During Modes 1 through 4 the manual actuation capability of SFAS will be maintained. During Modes 1 through 4, control room indication of normal and accident range radiation monitoring will be maintained in accordance with TS 3/4.3.3.1 and 3/4.4.6.1.

Under TS 3/4.6.1.7, requiring the CTMT purge and exhaust isolation valves to be closed with control power removed, and requiring an open CTMT purge and exhaust isolation valve to be closed with control power removed within 24 hours is more restrictive than the current Technical Specifications or ''The Improved Standard Technical Specifications for Babcock and Wilcox Plants,'' NUREG-1430, Revision 1. Under TS 3/4.9.4, the existing requirements already allow for the SFAS-initiated closure of the CTMT purge and exhaust isolation valves to be unavailable and the CTMT purge and exhaust system noble gas monitor used as an alternative means of achieving CTMT isolation. Further, it is noted that CTMT isolation is not credited in the accident analysis for the fuel handling accident. Therefore, these proposed changes do not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606

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

NRC Project Director: Gail H. Marcus

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

Date of application request: April 24, 1997, as supplemented by letters dated June 6, 1997, and June 27, 1997.

Description of amendment request: The amendment would revise Section 6.0 of the Technical Specifications to change the title ''Senior Vice President, Nuclear'' to ''Vice President and Chief Nuclear Officer.''

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

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

This proposed change does not involve any hardware or design changes, plant procedures, or administrative changes, other than a revision of title designation in documentation. Within the Union Electric organizational structure, the departments reporting to the former Senior Vice-President, Nuclear now report to the Vice President and Chief Nuclear Officer. The position of Vice-President and Chief Nuclear Officer now reports to the President & Chief Executive Officer of Union Electric, which is the same management level of reporting as the previous title, Senior Vice-President, Nuclear. This change has no impact on the probability or consequences of accidents previously evaluated in the Final Safety Report (FSAR).

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

This proposed change does not involve any hardware or design changes, plant procedures, or administrative changes, other than a revision of title designation in documentation. Within the Union Electric organizational structure, the departments reporting to the former Senior Vice-President, Nuclear now report to the Vice President and Chief Nuclear Officer. The position of Vice-President and Chief Nuclear Officer now reports to the President & Chief Executive Officer of Union Electric, which is the same management level of reporting as the previous title, Senior Vice-President, Nuclear. No new or different kind of accident is introduced by this purely administrative change to revise documentation to reflect current organizational titles.

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

The safety margins of the Technical Specifications are based on the actual plant design and are unaffected by this purely administrative change. This change merely updates the Technical Specifications to reflect the current organizational title for senior management of the Callaway Plant, and within the organizational structure of Union Electric. This change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on 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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

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: May 14, 1997

Description of amendment request: The proposed change will provide clarification to the testing and inspection requirements that each of the turbine control valves be cycled and movement verified through at least one complete cycle from the running position and revise the current wording in Surveillance Requirement 4.7.1.7.2.a for both units to clarify the testing and inspection methodology of the turbine control valves. Additionally, Technical Specification Bases Section 3/4.7.1.7 will be revised to clarify the testing requirements for the turbine governor control valves.

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:

Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

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

No new or unique accident precursors are introduced by these changes in surveillance requirements. The clarification for the turbine control valve testing and inspections do not change the design, operation, or failure modes of the valves and other components in the turbine overspeed protection system.

The verification of the operability of the turbine control valves will continue to provide adequate assurance that the turbine overspeed protection system will operate as designed, if needed. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previous[ly] evaluated.

Since the implementation of the proposed change to the surveillance requirements is to clarify the wording only, operation of the facilities with these proposed Technical Specifications does not create the possibility for any new or different kind of accident which has not already been evaluated in the Updated Final Safety Analysis Report (UFSAR).

The proposed wording changes to the Technical Specifications will not result in any physical alteration to any plant system, nor would there be a change in the method by which any safety-related system performs its function. The design and operation of the turbine overspeed protection and turbine control systems are not being changed. The proposed change merely represents a clarification to more specifically state current test requirements and test practice.

These changes do not change the design, operation, or failure modes of the valves and other components of the turbine overspeed protection system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes would not reduce the margin of safety as defined in the basis for any Technical Specifications. The design and operation of the turbine overspeed protection and turbine control systems are not being changed and the operability of the turbine control valves are being demonstrated in the same manner. In addition, the results of the accident analyses which are documented in the UFSAR continue to bound operation under the proposed changes, so that there is no safety margin reduction. Therefore, the proposed change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 23219

NRC Project Director: Gordon E. Edison, Acting

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

Date of amendment request: July 3, 1997

Description of amendment request: This license amendment request revises Technical Specification Section 5.3.1, Fuel Assemblies, to allow the use of an alternate zirconium based fuel cladding material, ZIRLO. Wolf Creek Nuclear Operating Corporation (WCNOC) is planning to insert Westinghouse fuel assemblies containing ZIRLO fuel rod cladding during the ninth refueling outage, which is currently scheduled to begin in October 1997. This request proposes to incorporate additional information, associated with the requested change, into Technical Specification 6.9.1.9, ''CORE OPERATING LIMITS REPORT (COLR).'' This revised submittal supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that were published on April 23, 1997 (62 FR 19839).

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

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

The methodologies used in the accident analysis remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Use of ZIRLO fuel cladding does not adversely affect fuel performance or impact nuclear design methodology. Therefore accident analyses are not impacted.

The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC approved methodologies. Other than the changes to the fuel assemblies, there are no physical changes to the plant associated with this technical specification change. A safety analysis will continue to be performed for each cycle to demonstrate compliance with all fuel safety design basis.

VANTAGE 5H with IFMs fuel assemblies with ZIRLO clad fuel rods meet the same fuel assembly and fuel rod design bases as other VANTAGE 5H with IFMs fuel assemblies. In addition, the 10 CFR 50.46 criteria are applied to the ZIRLO clad rods. The use of these fuel assemblies will not result in a change to the reload design and safety analysis limits. The clad material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO cladding improves corrosion performance and dimensional stability. No concerns have been identified with respect to the use of an assembly containing a combination of Zircaloy-4 and ZIRLO clad fuel rods. Since the dose predictions in the safety analyses are not sensitive to fuel rod cladding material, the radiological consequences of accidents previously evaluated in the safety analysis remain valid.

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

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

VANTAGE 5H with IFMs fuel assemblies with ZIRLO clad fuel rods satisfy the same design bases as those used for other VANTAGE 5H with IFMs fuel assemblies. All design and performance criteria continue to be met and no new failure mechanisms have been identified. Since the original design criteria are met, the ZIRLO clad fuel rods will not be an initiator for any new accident or malfunction of equipment important to safety. The ZIRLO cladding material offers improved corrosion resistance and structural integrity.

The proposed changes do not affect the design or operation of any system or component in the plant. The safety functions of the related structures, systems or components are not changed in any manner, nor is the reliability of any structure, system or component reduced. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure or system. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems and components are not affected, the proposed changes do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any accident or malfunction of equipment important to safety previously evaluated.

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

Use of ZIRLO cladding material does not change the VANTAGE 5H with IFMs reload design and safety limits. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core, the fuel assemblies will be evaluated using NRC approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects.

The use of Zircaloy-4, ZIRLO or stainless steel filler rods in fuel assemblies will not involve a significant reduction in the margin of safety because analyses using NRC approved methodologies will be performed for each configuration to demonstrate continued operation within the limits that assure acceptable plant response to accidents and transients. These analyses will be performed using NRC approved methods that have been approved for application to the fuel configuration.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

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

Date of amendment request: July 3, 1997

Description of amendment request: This license amendment request revises Definition 1.9, ''CORE ALTERATION.'' This change will more clearly define the types of components that constitute a core alteration when moved.

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

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

The probability of occurrence of a previously evaluated accident is not increased because this change to the definition of core alteration does not introduce any new potential accident initiating conditions. The proposed change will not affect any previously evaluated accident scenario. This proposed change will not affect any currently approved refueling-related operating activities. The consequences of an accident previously evaluated is not increased because the ability of containment to restrict the release of any fission product radioactivity to the environment will not be degraded by this change.

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

The proposed change does not affect any previously evaluated accident scenarios, nor does it create any new accident scenarios. The proposed change does not alter any of the currently-approved refueling operation activities, nor does it create any new refueling operating activities.

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

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

WCGS Technical Specification 3/4.9.1, Boron Concentration, specifies that K_{eff} will be maintained equal to or less than 0.95 during Operating Mode 6 with fuel in the vessel and the vessel head removed. The proposed change in the definition of core alteration will allow ''non-core'' components, such as cameras, lights, fuel inspection tools, etc., to be moved or manipulated in the vessel, with fuel in the vessel and the vessel head removed, without constituting a core alteration. This is acceptable because these types of components will have no effect on core reactivity, and will not affect reactor coolant system boron concentrations. Therefore, operations using these types of components will not adversely affect K_{eff} or the shutdown margin. Reactor subcriticality status is continuously monitored in the control room during Operating Mode 6, as specified in WCGS Technical Specification 3/4.9.2, Instrumentation.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

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

Date of amendment request: July 3, 1997

Description of amendment request: This license amendment request revises Surveillance Requirements 4.3.1.2 and 4.3.2.2 of Technical Specification (TS) 3/4.3.1, ''Reactor Trip System Instrumentation'' and TS 3/4.3.2, ''Engineered Safety Features Actuation System Instrumentation'' and associated Bases to indicate that the total response time will be determined based on the results of WCAP-13632-P-A Revision 2, ''Elimination of Pressure Sensor Response Time Testing Requirements.''

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

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

The same RTS [Reactor Trip System] and ESFAS [Engineered Safety Features Actuation System] instrumentation is being used. The time response allocations/modeling assumptions in the Updated Safety Analysis Report Chapter 15 analyses are still the same, only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. The proposed change will not affect the probability of any event initiators, nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of, nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no hardware changes, nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not alter the normal method of plant operation. No transmitter performance requirements will be affected. This change does not alter the performance of the pressure and differential pressure transmitters used in the plant protection systems. All sensors will still have response times verified by test before placing the sensors in operational service, and after any maintenance that could affect response time. Changing the method of periodically verifying instrument response for certain sensors (assuring equipment operability) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the sensor response characteristic. No new transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change does not affect the acceptance criteria for any analyzed event. This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and differential pressure sensors is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within

that defined in the safety analysis, since calibration tests will detect any degradation which might significantly affect sensor response time. There will be no effect on the manner in which safety limits or limiting safety system settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

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.

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

Date of application for amendment: January 27, 1997, as supplemented May 16, 1997.

Brief description of amendment: The amendment revised the Technical Specifications to permit control rod misalignment of plus or minus 18 steps when the core power is less than or equal to 85% of rated thermal power (RTP) and plus or minus 12 steps above 85% RTP.

Date of publication of individual notice in **Federal Register:** June 19, 1997 (62 FR 33445)

Expiration date of individual notice: July 21, 1997

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

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.

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: January 20, 1997

Brief description of amendments: The amendments revise Technical Specification (TS) 3.6.3, ''Containment Isolation Valves,'' to reflect modifications associated with steam generator replacement for Unit 1 of each station. TS Table 3.6-1, ''Containment Isolation Valves,'' will be modified to reflect the deletion of feedwater bypass valves and reassignment of certain isolation valves to different containment penetrations. TS pages for Unit 2 of each station are affected because Units 1 and 2 share common TS pages.

Date of issuance: : July 10, 1997Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 91, 90, 84, and 83

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:** March 12, 1997 (62 FR 11489). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 10, 1997. No 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 No. 50-010, Dresden Nuclear Generating Station, Unit 1, Grundy County, Illinois

Date of application for amendment: October 23, 1996, as supplemented November 25, 1996, and June 5, 1997.

Brief description of amendment: The amendment replaces the Appendix A Technical Specifications of License DPR-2 in their entirety. The amendment revises the Dresden 1 Technical Specifications (TS) to the same format as the Dresden Nuclear Power Station, Units 2 and 3 (Dresden 2/3) Technical Specification Upgrade Program (TSUP).

Date of issuance: July 8, 1997

Effective date: July 8, 1997

Amendment No.: 39

Facility Operating License No. DPR-2: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4343). The November 25, 1996, and June 5, 1997, submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 8, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: January 20, 1997

Brief description of amendments: The amendments revise the Technical Specifications for various instruments which have alarm or indication functions. The amendments relocate surveillance requirements for selected instrumentation from Technical Specifications to licensee controlled documents or replace selected surveillance requirements with those more appropriate to the associated LCOs. In addition, the amendments add an action statement related to the automatic depressurization system accumulator backup compressed gas system and delete action statements related to suppression chamber water level instrumentation.

Date of issuance: July 16, 1997

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: 118 and 103

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

Date of initial notice in **Federal Register:** February 26, 1997 (62 FR 8795) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 16, 1997. No significant hazards consideration comments received: No.

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

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

Date of amendment request: August 15, 1996, as supplemented by letters dated October 31, 1996, and May 29, 1997.

Brief description of amendments: The amendments removed a requirement for performance of a surveillance incorporating a high toxic gas test signal.

Date of issuance: July 17, 1997

Effective date: July 17, 1997, to be implemented within 30 days.

Amendment Nos.: Unit 1 - Amendment No. 88; Unit 2 - Amendment No. 75

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

Date of initial notice in **Federal Register:** September 25, 1996 (61 FR 50344) The additional information contained in the supplemental letters dated October 31, 1996, and May 29, 1997, were clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior

College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488

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

Date of application for amendment: April 14, 1997

Brief description of amendment: Technical Specification 3.4.9.3 requires, in part, that two residual heat removal suction relief valves be operable to protect the reactor coolant system from overpressurization when any reactor coolant system cold leg is less than 350 degrees. The amendment revises the setpoint of the residual heat removal suction relief valves.

Date of issuance: July 10, 1997

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

Amendment No.: 143

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

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30634) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 10, 1997. No significant hazards consideration comments received: No.

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

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: June 10, 1996, as supplemented July 25, 1996

Brief description of amendments: These amendments change the differential temperature Technical Specifications allowable values and trip setpoints for the reactor water cleanup system penetration room steam leak detection function.

Date of issuance: June 26, 1997

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

Amendment Nos.: 166 and 140

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

Date of initial notice in **Federal Register:** December 4, 1996 (61 FR 64389) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 26, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 23, 1996, as supplemented February 26, 1997, May 12, 1997, June 16, 1997, and July 2, 1997 and July 11, 1997.

Brief description of amendment: The amendment changes the Technical Specifications to allow the use of VANTAGE+ fuel for cycle 10.

Date of issuance: July 15, 1997

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

Amendment No.: 175

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 12, 1997 (62 FR 6578). The February 26, 1997, May 12, 1997, and June 16, 1997, July 2, 1997 and July 11, 1997, letters provided information that did not change the initial no proposed significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 15, 1997. No significant hazards consideration comments received: No.

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

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: March 31, 1997

Brief description of amendment: This amendment changes Hope Creek Technical Specification Section 3.6.5.3.2, ''Filtration, Recirculation and Ventilation System (FRVS),'' to provide an appropriate Limiting Condition for Operation and ACTION Statement that reflects the design basis for the FRVS.

Date of issuance: July 9, 1997

Effective date: July 9, 1997, to be implemented within 60 days

Amendment No.: 99

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27798) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 9, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070

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

Date of application for amendments: June 18, 1996, as supplemented August 19, 1996, April 28, 1997, and June 11, 1997

Brief description of amendments: The amendments change Technical Specification (TS) 5.2.2, ''Design Pressure and Temperature,'' by adding design parameters for Main Steam Line Break (MSLB). The MSLB analysis results in a higher containment air temperature than the value that was in TS 5.2.2 prior to the issuance of these amendments.

Date of issuance: July 17, 1997

Effective date: July 17, 1997

Amendment Nos.: 198 and 181

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

Date of initial notice in **Federal Register:** July 17, 1996 (61 FR 37302) The supplemental letters did not change the original no significant hazards consideration determination nor the Federal Register notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 1997. No significant hazards consideration comments received: No

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

Tennessee Valley Authority, Docket Nos. 50-259, 50-260, and 50-296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: August 30, 1996 (TS 380)

Brief description of amendment: The amendments remove License Condition 2.C.(3) regarding thermal water quality limits.

Date of issuance: July 8, 1997

Effective Date: Effective as of the date of issuance.

Amendment Nos.: 232, 248 and 208

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revise the license.

Date of initial notice in **Federal Register:** September 25, 1996 (61 FR

50347) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 8, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 35611

Tenessee 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 22, 1996 (TS 96-08)

Brief description of amendments: The amendments change the Technical Specifications (TS) by eliminating the emergency diesel generator accelerated testing and special reporting requirements TS 4.8.1.1.2.a in accordance with NRC Generic Letter 94-01.

Date of issuance: : July 14, 1997

Effective date: July 14, 1997

Amendment Nos.: 226 and 217

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

Date of initial notice in **Federal Register:** October 9, 1996 (61 FR 52969) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 14, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: December 17, 1996

Brief description of amendments: The proposed changes will allow one of the two service water loops to be isolated from the component cooling water heat exchangers (CCHXs) during power operation in order to refurbish sections of the isolated service water headers. The proposed temporary changes will be valid for two periods of up to 35 days each for implementation of the service water upgrades associated with the repair of the sections of the 24-inch service water supply and return piping to/from the CCHXs.

Date of issuance: July 17, 1997

Effective date: July 17, 1997

Amendment Nos.: 205 and 186

Facility Operating License Nos. NPF-4 and NPF-7:. These amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 12, 1997 (62 FR 6580) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498Virginia Electric and Power Company, et al., Docket Nos. 50-280 and 50-281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: November 26, 1997

Brief description of amendments: These amendments revise the Technical Specifications (TSs) to eliminate the records retention requirements from Section 6.5 of the TSs. The relocation of those requirements to the Operational Quality Assurance program, contained in the Final Safety Analysis Report, has been completed.

Date of issuance: July 15, 1997

Effective date: July 15, 1997

Amendment Nos.: 211 and 211

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

Date of initial notice in **Federal Register:** March 26, 1997 (62 FR 14472) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 15, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 3, 1997, and March 18, 1997

Brief description of amendments: These amendments revise the Technical Specifications to eliminate the inconsistency between the current approved Inservice Inspection Program and ASME Code (1989 Edition) and the Surry Technical Specifications (TS) as required by 10 CFR 50.55a(g)95)(ii).

Date of issuance: July 15, 1997

Effective date: July 15, 1997

Amendment Nos.: 212 and 212

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

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 17242) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 15, 1997. No significant hazards consideration comments received: No.

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

Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: May 20, 1997, as supplemented by letters dated June 6, 1997, and July 3, 1997. Additional information was also received by letters dated June 12, June 20, and June 25, 1997.

Brief description of amendment: The amendment modifies the Technical Specifications (TS) for the minimum critical power ratio (MCPR) safety limit in TS 2.1.1.2 for ATRIUM 9X9 fuel. This change is effective for Cycle 13 operation only.

Date of issuance: July 3, 1997

Effective date: July 3, 1997, to be implemented within 30 days from the date of issuance.

Amendment No.: 151

Facility Operating License No. NPF-21: The amendment revised the Technical Specifications and operating license.

Date of initial notice in **Federal Register:** May 29, 1997 (62 FR 29160). The June 12, June 20, June 25, and July 3, 1997, submittals provided clarifying information which did not affect the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 3, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352

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

Date of application for amendments: September 30, 1996 (TSCR-192), as supplemented on November 26 and December 12, 1996, February 13, March 5, April 2, April 16, May 9, June 3, June 13 (two letters), and June 25, 1997

Brief description of amendments: These amendments revise Technical Specification (TS) 15.3.3, ''Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray,'' to incorporate allowed outage times similar to those contained in NUREG-1431, Revision 1, ''Westinghouse Owner's Group Improved Standard Technical

Specifications,'' and modify the operability requirements for the service water and component cooling water systems. TS 15.3.7, ''Auxiliary Electrical Systems,'' was revised to reflect the changes to the service water system operability requirements. These changes ensure that TS requirements are the ''lowest functional capability or performance levels of equipment required for safe operation of the facility,'' as defined in 10 CFR 50.36(c)(2), ''Limiting Conditions for Operation.'' Additionally, the amendments change TS 15.3.12, ''Control Room Emergency Filtration,'' to revise charcoal filtration efficiencies and to include a specific testing standard, and TS 15.5.2, ''Containment,'' to revise the design heat removal capability of the containment fan coolers.

Date of issuance: July 9, 1997

Effective date: July 9, 1997, with full implementation prior to restart of Unit 2 and Unit 1 and no later 45 days from the date of issuance. Implementation includes incorporating changes to TS requirements for the service water system, component cooling water systems, and control room ventilating system as detailed in an application dated September 30, 1996, as supplemented on November 26 and December 12, 1996, February 13, March 5, April 2, April 16, May 9, June 3, June 13 (two), and June 25, 1997, and evaluated in the staff's safety evaluation dated July 9, 1997. These amendments are authorized contingent on compliance to commitments provided by the licensee, to meet the dose limits associated with Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion (GDC) 19 by: (1) submitting a license amendment application including supporting analyses and evaluations by February 27, 1998, that contains the proposed methods for compliance with GDC 19 dose limits under accident conditions based on system design and without reliance on the use of potassium iodide and/or self contained breathing apparatus, and (2) implementing the proposed changes within 2 years of the date that NRC approval for the proposed license amendment is granted. Additionally, these amendments are authorized contingent on compliance to commitments provided by the licensee, to operate Point Beach Nuclear Plant in accordance with its service water system analyses and approved procedures. Specifically, each unit will utilize only one component cooling water (CCW) heat exchanger until such time that analyses are completed and the service water system reconfigured as necessary to allow operation of one or both units with two heat exchangers in service. If two CCW heat exchangers are required in one or both units for maintaining acceptable CCW temperature prior to completion of necessary analyses to allow operation in the required configuration, the service water system will be considered in an unanalyzed condition, declared inoperable and action taken as specified by TS 15.3.0.B except for short periods of time as necessary to effect procedurally controlled changes in system lineups and unit operating conditions.

Amendment Nos.: 174 and 178

Facility Operating License Nos. DPR-24 and DPR-27: Amendments revised the Licenses and Technical Specifications. Public comments requested as to proposed no significant hazards considerations (NSHC): Yes (61 FR 58905 dated November 19, 1996; 62 FR 17244 dated April 9, 1997; and 62 FR 31636 dated June 10, 1997). No comments have been received. The June 10, 1997, notice also provided for an opportunity to request a hearing by July 10, 1997, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendments. The June 13 and June 25, 1997, submittals provided clarifying information within the scope of the application and did not affect the staff's previous no significant hazards considerations determinations. The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of no significant hazards considerations are contained in a Safety Evaluation dated July 9, 1997.

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

Local Public Document Room location: The Lester Public Library 1001 Adams Street, Two Rivers, WI 54241

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

Date of amendment request: April 23, 1997

Brief description of amendment: This amendment allows the service air and breathing air containment penetrations to remain open under administrative control during periods of core alterations or movement of irradiated fuel inside containment.

Date of issuance: July 11, 1997

Effective date: July 11, 1997, to be implemented within 30 days from the date of issuance.

Amendment No.: 107

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

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30648) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 1997. 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

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By August 29, 1997, 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, Attention: 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. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, and to the attorney for the licensee.

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

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

Date of amendment request: July 10, 1997

Brief description of amendment: The amendment changes the Appendix A Technical Specifications by deleting the requirements of Surveillance Requirements (SR) 4.8.1.1.2.h.2 for the diesel fuel oil system. This change will result in testing of the diesel fuel oil system in accordance with ASME Code Section XI requirements.

Date of issuance: July 11, 1997

Effective date: July 11, 1997, with full implementation within 30 days.

Amendment No: 132

Facility Operating License No. NPF-38: Amendment revises the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated July 11, 1997.

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

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

NRC Acting Project Director: James Clifford, Acting

Dated at Rockville, Maryland, this 23rd day of July 1997.

For The Nuclear Regulatory Commission

Jack W. Roe,

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

[Doc. 97–19910 Filed 7–29–97; 8:45 am]

BILLING CODE 7590–01–F

POSTAL RATE COMMISSION

[Docket No. A97–25, Order No. 1187]

In the Matter of: Webster Crossing, New York 14584, (Eleanor Wong, et al., Petitioners); Notice and Order Accepting Appeal and Establishing Procedural Schedule UNDER 39 U.S.C. § 404(b)(5)

Issued July 24, 1997.

Docket Number: A97–25.

Name of Affected Post Office: Webster Crossing, New York 14584.

Name(s) of Petitioner(s): Eleanor Wong, *et al.*

Type of Determination: Closing.

Date of Filing of Appeal Papers: July 18, 1997.

Categories of Issues Apparently Raised:

1. Effect on the community [39 U.S.C. 404(b)(2)(A)].

2. Effect on postal services [39 U.S.C. 404(b)(2)(C)].

After the Postal Service files the administrative record and the Commission reviews it, the Commission may find that there are more legal issues than those set forth above. Or, the Commission may find that the Postal Service's determination disposes of one or more of those issues.

The Postal Reorganization Act requires that the Commission issue its decision within 120 days from the date this appeal was filed (39 U.S.C. 404(b)(5)). In the interest of expedition, in light of the 120-day decision schedule, the Commission may request the Postal Service to submit memoranda of law on any appropriate issue. If requested, such memoranda will be due 20 days from the issuance of the request and the Postal Service shall serve a copy of its memoranda on the petitioners. The Postal Service may incorporate by reference in its briefs or motions, any arguments presented in memoranda it previously filed in this docket. If necessary, the Commission also may ask petitioners or the Postal Service for more information. The Commission orders:

(a) The Postal Service shall file the record in this appeal by August 1, 1997.

(b) The Secretary of the Postal Rate Commission shall publish this Notice and Order and Procedural Schedule in the **Federal Register**.

By the Commission.

Cyril J. Pittack,

Acting Secretary.

Appendix

July 18, 1997—Filing of Appeal letter.
July 24, 1997—Commission Notice and Order of Filing of Appeal.
August 12, 1997—Last day of filing of petitions to intervene [see 39 CFR 3001.111(b)].
August 22, 1997—Petitioners' Participant Statement or Initial Brief [see 39 CFR 3001.115 (a) and (b)].
September 11, 1997—Postal Service's Answering Brief [see 39 CFR 3001.115(c)].
September 26, 1997—Petitioners' Reply Brief should Petitioner choose to file one [see 39 CFR 3001.115(d)].
October 3, 1997—Deadline for motions by any party requesting oral argument. The Commission will schedule oral argument only when it is a necessary addition to the written filings [see 39 CFR 3001.116].
November 15, 1997—Expiration of the Commission's 120-day decisional schedule [see 39 U.S.C. 404(b)(5)].

[FR Doc. 97–20014 Filed 7–29–97; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–22762; File No. 812–10676]

Oppenheimer & Co., L.P., et al.

July 24, 1997.

AGENCY: Securities and Exchange Commission (''SEC'').

ACTION: Notice of application for exemption under the Investment Company Act of 1940 (''Act'').

Applicants: Oppenheimer & Co., L.P. (''Opco''), Oppenheimer Group, Inc. (''Opgroup''), Oppenheimer Financial Corp. (''Opfin'') (collectively, the ''Oppenheimer Applicants''), The Emerging Markets Income Fund Inc. (''Emerging Market''), The Emerging Markets Income Fund II Inc. (''Emerging Market II''), The Emerging Markets Floating Rate Fund Inc. (''Emerging floating Rate''), Global Partners Income Fund Inc. (''Global Partners''), Municipal Partners Fund Inc. (''Municipal Partners''), Municipal Partners Fund II Inc. (''Municipal Partners II''), The Enterprise Group of Funds, Inc. (''Enterprise Fund''), Enterprise Accumulation Trust (''Enterprise Trust''), WNL Series Trust (''WNL''), Endeavor Series Trust (''Endeavor''), Penn Series Funds, Inc. (''Penn Fund''), The Preferred Group of Mutual Funds (''Preferred''), Select Advisors Portfolios (''Select Portfolios''), Select Advisors Variable Insurance Trust (''Select Trust''), Select Advisors Trust A (''Select A''), and Select Advisors Trust C (''Select C'') (collectively, the ''Companies'').

Relevant Act Sections: Order requested under section 6(c) for an exemption from section 15(f)(1)(A).

Summary of Application: Applicants request an exemption from section 15(f)(1)(A) in connection with the proposed change in control of Oppenheimer Capital (''Opcapital''), Opcap Advisors (''Opcap''), and Advantage Advisers, Inc. (''Advantage,'' collectively with Opcapital and Opcap, the ''Advisers''), each of which acts as investment adviser or subadviser to one or more of the Companies. Without the requested exemption, the Companies would have to reconstitute their boards of directors (''Boards'') to meet the 75