VIII.A and VIII.B will be reviewed and resolved by the Director, Office for Analysis and Evaluation of Operational Data.

D. The NRC's General Counsel has the final authority to provide legal interpretation of the Commission's regulations.

IX. Effective Date

This Agreement will take effect after it has been signed by both parties.

X. Duration

A formal review, not less than 1 year after the effective date, will be performed by the NRC to evaluate implementation of the Agreement and resolve any problems identified. This Agreement will be subject to periodic reviews and may be amended or modified upon written agreement by both parties, and may be terminated upon 30 days written notice by either party.

XI. Separability

If any provision(s) of this Agreement, or the application of any provision(s) to any person or circumstances is held invalid, the remainder of this Agreement and the application of such provisions to other persons or circumstances will not be affected.

For the U.S. Nuclear Regulatory Commission,

James M. Taylor,

Executive Director for Operations.

For the State of Louisiana.

Dated: October 31, 1996.

Gus Von Bodungen,

Assistant Secretary, Office of Air Quality and Radiation Protection, Department of Environmental Quality.

[FR Doc. 96–30902 Filed 12–3–96; 8:45 am]

BILLING CODE 7590–01–M

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 November 8, 1996, through November 21, 1996. The last biweekly notice was published on November 19, 1996.

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

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

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

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

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

By January 3, 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. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

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: October 31, 1996

Description of amendment request: The proposed change would revise the maximum allowable water temperature as measured at the respective intake structures from 95°F to 94°F and will increase the minimum main reservoir level from 205.7 feet mean sea level to 215 feet mean sea level in Technical Specification (TS) 3/4.7.5, Ultimate Heat Sink.

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

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

Since the proposed change does not affect the operation of any accident initiating systems, the probability of occurrence of an accident previously evaluated will not increase. Also, none of the proposed changes will cause plant systems to operate outside their design limits or create the likelihood of a radioactive release. Therefore, there would be no increase in the 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.

No new component or system level interactions will be created by the proposed change in ultimate heat sink level and temperature, and no design limits will be exceeded. This change to [Technical] Specification 3/4.7.5 is more conservative than the current Specification limits and will serve only to restrict operation to a higher reservoir level and lower temperature than was previously allowed. 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 will establish a more conservative minimum main reservoir level such that safety-related heat exchangers served by Emergency Service Water will continue to remove their design-basis accident heat loads. Establishing a higher minimum reservoir level, combined with a more conservative reservoir temperature assumption, will involve an increase in the margin of safety. Also, the proposed change in maximum reservoir temperature from 95°F to 94°F will not result in any reduction in the margin of safety. A maximum pre-accident initial water temperature of 94°F is necessary to yield a post-accident (30-day) calculated maximum inlet temperature less than or equal to the design basis temperature of 95°F. 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: Mark Reinhart, Acting

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

Date of amendment request: November 4, 1996

Description of amendment request: The proposed amendments would eliminate from the Technical Specifications, Section 4.7.13.1, the ''during shutdown'' restriction pertaining to the 18-month Standby Shutdown System (SSS) diesel generator inspection. Unlike Catawba Nuclear Station, many nuclear plants do not have an SSS facility and associated diesel generator. The requirements in the Technical Specifications for the SSS diesel generator (shared between both units) were patterned after similar requirements for the emergency diesel generators. The current wording requires that both units be shut down to perform the subject inspection.

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

... The standard for determining that a Technical Specification amendment request involves no significant hazards considerations requires that operation of the facility in accordance with the requested amendment will 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 the margin of safety.

Criterion 1

The proposed amendment seeks to change the surveillance requirements to allow the SSS DG [diesel generator] periodic inspection with one or both units on line. The surveillance can be safely completed as proposed without affecting unit operation. The equipment would not be removed from service for a time that would exceed the current Limiting Condition For Operation or the appropriate action statement would be entered. The probability or consequences of any accident previously evaluated will not be significantly increased because the removal of the SSS DG from service can be safely performed while one or both units are operating.

Criterion 2

The proposed amendment change does not change any actual surveillance requirements. The change simply allows the 18 month SSS DG inspection to be performed at different unit conditions. The performance of the surveillance with the units operating do not require any new component configurations that would reduce the ability of any equipment to mitigate an accident. The station is not degraded beyond that which has been previously evaluated. Therefore the proposed change does not create the possibility of a new or different kind of accident.

Criterion 3

The allowed outage time for the SSS DG, as specified by the Limiting Condition For Operation, defines the required margin of safety for equipment operability. Removing the SSS DG from service for periodic inspection and returning it to service within the allowed outage time 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 proposed amendments involve no significant hazards consideration.

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: October 30, 1996

Description of amendment request: The proposed changes would (1) completely rewrite Technical Specification (TS) 4.4.2 to incorporate a prestressed concrete containment surveillance program that is consistent with Regulatory Guide 1.35, (2) modify TS 3.6.7 by establishing new Limiting Conditions for Operation and required actions related to the structural integrity of the reactor buildings, (3) incorporate an editorial change to TS 6.6.3 to reference the relocated tendon surveillance reporting requirements, and (4) modify TS 3.6.7 Bases to describe the Reactor Building post-tensioning TS.

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

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

No. The proposed amendment to Oconee Technical Specifications involves the implementation of an enhanced surveillance program for the reactor building prestressed concrete containment and the assurance of appropriate station response to abnormal degradation of the containment structure. The proposed change will move Oconee into a surveillance program which is consistent with accepted industry practice and a published NRC regulatory position. The adoption of Regulatory Guide 1.35 as a basis for the periodic inspection of the reactor building prestressed concrete containment and clearly defined station response to any indication of structural deterioration will assure acquisition of sufficient data to demonstrate that structural integrity is maintained and, if necessary, appropriate compensatory action(s) are taken. By assuring that any adverse trends in the behavior of the prestressed concrete containment are identified and acted upon in a timely manner, this change does not increase the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment to Oconee Technical Specifications involves the implementation of an enhanced surveillance program for the reactor building prestressed concrete containment and the assurance of appropriate station response to abnormal degradation of the containment structure. By adopting Regulatory Guide 1.35 as a basis for the surveillance inspection program for the reactor building prestressed concrete containment and clearly defining required station response to any indication of structural deterioration, sufficient data will be obtained to demonstrate that structural integrity is being maintained and that any adverse behavioral trends are identified and acted upon in a timely manner. Therefore, the proposed amendment does not create the possibility of any type of accident: new, different or previously evaluated.

3) Will the change involve a significant reduction in a margin of safety?

No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed Technical Specifications amendment will move Oconee into a surveillance program which is consistent with accepted industry practice and a published regulatory position. By ensuring more timely identification of, and response to, any adverse trend in the behavior of the reactor building prestressed concrete containment, continued maintenance of the structural integrity is enhanced. Therefore, the ability of the containment structure to perform the intended function of protecting the public from radiation dose is further assured, and no reduction in any existing margin of safety will occur.

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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: September 9, 1996

Description of amendment request: The proposed amendment would modify the design features section (Section 5.0) of the Technical Specifications (TSs) to make the design features section consistent with the four criteria specified in the Commission's Policy Statement on TSs (58 FR 39132) and with the guidance provided in the NRC's Standard Technical Specifications, Westinghouse Plants (NUREG-1431, Revision 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. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change reduces the content of the technical specification (TS) design feature section consistent with the Improved Standard Technical Specifications (ISTS) of NUREG-1431. The information that has been removed is also contained in the UFSAR [Updated Final Safety Analysis Report] or offsite dose calculation manual (ODCM); therefore, duplication of the information is eliminated to improve the use of the TS. Because the information removed from the TS is maintained in the UFSAR or ODCM where changes are controlled in accordance with regulatory requirements, there is no reduction in commitment and adequate control is provided. Elimination of information from the design feature section of the TS which duplicates information in the UFSAR enhances the usability of the TS without reducing commitments. These changes clarify and improve the understanding and readability of the TS. Since the requirements remain the same, these changes only affect the method of presentation and would not affect possible initiating events for accidents previously evaluated or any system functional requirement. Therefore, the proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The relocation of existing requirements, the elimination of requirements which duplicate existing information, and making administrative improvements are all changes that are administrative in nature. The proposed changes will not affect any plant system or structure, not [nor] will they affect any system functional or operability requirements. Consequently, no new failure modes are introduced as a result of the proposed changes. The proposed changes are consistent with the ISTS, for the most part, as plant-specific information is included in this section. Therefore, the proposed change will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The proposed changes are administrative in nature in that no change to the design features of the facility are being made. The design features section is being reformatted to be consistent, for the most part, with the ISTS. The proposed changes do not affect the UFSAR design bases, accident assumptions, or technical specification bases. In addition, the proposed changes do not affect release limits, monitoring equipment or practices. 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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001

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

NRC Project Director: John F. Stolz

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

Date of amendment request: October 24, 1996

Description of amendment request: The proposed amendment would revise the technical specifications to remove accelerated testing requirements for the standby diesel generators. The changes implement the provisions of Generic Letter (GL) 94-01, ''Removal of Accelerated Testing and Special Reporting Requirements For Diesel Generators'', dated May 31, 1994.

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

This change will provide flexibility to structure the standby diesel generator maintenance program based on the risk significance of the structures, systems, and components that are within the scope of the Maintenance Rule. The removal of the diesel generator accelerated testing is acceptable as the maintenance rule applies site and system specific performance criteria to monitor diesel generator performance. This criteria includes a running availability and reliability goal as well as specific goals to monitor maintenance preventable functional failures. The performance criteria for the diesel generator reliability and unavailability established by the maintenance rule and the causal determinations and corrective actions required for maintenance preventable functional failures are considered to be an acceptable method for monitoring diesel generator performance.

The proposed change has no effect on the probability of the initiation of an accident, because the emergency diesel generators do not serve as the initiator of any event. Additionally, as diesel generator performance will continue to be assured by the maintenance rule, the proposed changes do not affect the ability to mitigate the consequences of an accident previously evaluated. The changes do not impact the diesel's design sources, operating characteristics, system functions, or system interrelationships. The failure mechanisms for the accidents previously analyzed are not affected and no additional failure modes are created that could cause an accident that has been previously evaluated. Since the diesel generator performance and reliability will continue to be assured by the maintenance rule, the proposed changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change does not involve a change to the plant design or operation. As a result, the proposed changes does not affect any of the parameters or conditions that could contribute to the initiation of any accidents. The proposed changes only affect the methods used to monitor and assure diesel generator performance. The performance criteria for both the diesel generator reliability and unavailability established by the maintenance rule, and the casual determinations and corrective actions required for maintenance preventable functional failures, is considered by GL 94-01 to be an acceptable method for monitoring diesel generator performance.

No [system, structure, or component] SSC, method of operating, or system interface is altered by this change. The changes do not impact the diesel's design sources, operating characteristics, system functions, or system interrelationships. The failure mechanisms for the accidents are not affected, and no additional failure modes are created. Because the diesel generator performance and reliability will continue to be assured by the maintenance rule, the proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The request does not involve a significant reduction in a margin to safety.

The proposed changes only affect the methods used to monitor and assure diesel generator performance and reliability. The performance criteria for both the diesel generator reliability and unavailability established by the maintenance rule, and the casual determinations and corrective actions required for maintenance preventable functional failures, is considered by GL 94-01 to be an acceptable method for monitoring diesel generator performance. No margin to safety as defined in the basis for any technical specification is impacted by these changes. This change does not impact any uncertainty in the design, construction, or operation of any SSC. Diesel generator response to accident initiators is unchanged. No SSC, method of operating, or system interface is altered by this change. The changes do not impact the diesel's design sources, operating characteristics, system functions, or system interrelationships. Because the diesel generator performance and reliability will continue to be assured by the maintenance rule, the proposed changes cannot involve a significant reduction in the margin to safety.

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

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

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

Date of amendment request: November 6, 1996

Description of amendment request: The proposed amendment would revise the River Bend Station (RBS) Fire Hazards Analysis Report and Safety Analysis Report to allow a deviation from 10 CFR Part 50, Appendix R, Section III.G.2.c with respect to the requirement for an area wide automatic fire suppression system in Fire Area C-16. The deviation would allow a 1-hour barrier to separate redundant trains of post fire safe shutdown equipment within the fire area and partial sprays on the protected train.

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 request does not involve an increase in the probability or consequences of an accident previously evaluated.

The event of concern is a fire in Fire Area C-16. The low fire loading and minimal concentration of exposed combustible material in Fire Area C-16 would limit fire spread. However, for this scenario, all unprotected equipment in Fire Area C-16 will be assumed lost. Fire Area C-16 contains cables for both Division I and Division II components required for post fire safe shutdown. The loss of both divisions of cables could preclude the ability of the plant to achieve post fire safe shutdown. Protection of the required Division II cables in a 1-hour fire barrier in conjunction with a partial area, automatic suppression system installed above and below the protected trays will ensure that post fire safe shutdown can be achieved.

In summary, the probability of a fire occurring in Fire Area C-16 is not affected. However, if a fire were to occur in Fire Area C-16 which caused the loss of Division I powered components, Division II powered components, by virtue of the 1-hour fire barrier and partial area, automatic suppression system, would remain available. The low fire loading and minimal concentration of exposed combustible material in Fire Area C-16 would limit fire spread. The proposed fire protection scheme provides a level of protection commensurate with the original design. Therefore, this request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Fire Area C-16 will be protected by a partial area, automatic suppression system installed above and below the protected cable trays. Fire suppression systems are generally used to limit fire spread, once the heat of the fire opens thermally sensitive sprinklers. The low fire loading and minimal concentration of exposed combustible material in Fire Area C-16 would aid in limiting fire spread, and would also limit the severity of any plausible fire. The previous analysis assumed all Division I components and cables in the area would be lost, and that the installed fire barrier would adequately protect the Division II cables routed through C-16. The required Division II cables will be enclosed in a 1-hour fire barrier with a partial area, automatic suppression system. These features provide a level of protection commensurate with that of the previous design. In addition, the total combustible loading in the area results in a maximum theoretical worst case fire duration of less than 1-hour.

In summary, if a fire were to occur in Fire Area C-16 which caused the loss of Division I powered components, post fire safe shutdown could still be achieved using Division II. Therefore, this request does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

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

In this case, the margin of safety is implicit rather than being explicitly expressed as a numerical value. An implicit margin of safety involves conditions for NRC acceptance. Since the RBS Technical Specification Bases do not specifically address a margin of safety for fire protection, the SAR [Safety Analysis Report], the NRC's Safety Evaluation Report (SER), and appropriate other licensing basis documents were reviewed to determine if the proposed change would result in a reduction in a margin of safety. As stated, in part, in Attachment 4 to NPF-47 [Facility Operating License; NPF-47]:

EOI [Entergy Operations, Inc.] shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment 22 and as approved in the SER dated May 1984 and Supplement 3 dated August 1985 subject to provisions 2 and 3 ....

As discussed in the Reason for Request, SSER [Supplemental Safety Evaluation Report] 3 dated August 1985 states, in part:

On the basis of its evaluation the staff finds that the applicant's fire protection program with approved deviations is in conformance with the guidelines of BTP CMEB [branch technical position, Chemical Materials and Engineering Branch] 9.5-1, [S]sections III.G, III.J, and III.O of Appendix R to 10 CFR [Part] 50, and GDC [General Design Criteria] 3, and is, therefore, acceptable.

Thus, the margin of safety in this case can be defined as conformance with the specified fire protection guidelines.

10 CFR [Part] 50, Appendix R, Section III.G.2, requires, in part, that redundant trains of post fire safe shutdown equipment located in the same fire area be separated by a 1-hour fire barrier and, in addition, that fire detection and an automatic fire suppression system be installed in the are under consideration. Since Fire Area C-16 will have a partial area, automatic suppression system, this fire area would deviate from the requirements of 10 CFR [Part] 50, Appendix R, Section III.G.2.c. However, as discussed previously, the installed partial area, automatic suppression system, the low fire loading and minimal amount of exposed combustibles compensate for the lack of a total, area wide, automatic fire suppression

system. There is no adverse impact on the ability to achieve and maintain post fire safe shutdown. Therefore, this request does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

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

Dates of amendment requests: October 28, 1996 (Two letters)

Description of amendment request: The licensee proposed to change the St. Lucie Units 1 and 2 Technical Specifications (TS) to implement 10 CFR 50, Appendix J, Option B, for containment leakage testing by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program.'' Changes include relocating the details for containment testing to the ''containment leakage rate testing program'' and adding the requirements of the containment leakage rate testing program to TS 6.8.4, which describes facility programs. Changes are also proposed to remove Tables 3.6-1, ''Containment Leakage Paths,'' and 3.6-2, ''Containment Isolation Valves'' from TS and relocate the information to plant procedures.

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

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

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated due to the following reasons:

a)These proposed changes are all consistent with NRC requirements and guidance for implementation of 10 CFR 50, Appendix J, Option B, except for the use of Bechtel Topical Report BN-TOP-1 for type A testing. BN-TOP-1 has been previously approved for use in accordance with 10 CFR 50 appendix J.

b) Based on industry and NRC evaluations performed in support of developing Option B, these changes potentially result in a minor increase in the consequences of an accident previously evaluated due to the increased testing intervals. However, the proposed changes do not result in an increase in the core damage frequency since the containment system is used for mitigation purposes only.

c) These changes are expected to result in increased attention to components with poor leakage test history as part of the performance-based nature of Option B, such that the marginally increased consequences from the expanded testing intervals may be further reduced or negated.

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

The use of the modified specifications can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the implementation of a performance-based program for containment leakage rate testing, since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The operating limits and functional capabilities of the affected systems, structures, and components are basically unchanged by the proposed amendments. The increase in intervals between leak-test surveillances will not significantly reduce the margin of safety as shown by findings in NUREG 1493, ''Performance-Based Containment Leak-Test Program'', which was based on implementation of the performance-based testing of Option B.

Therefore these changes do not involve a significant reduction in the margin of safety.The NRC staff has reviewed the licensee's analysis and, based on thisreview, 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, 11770 US Highway 1, North Palm Beach, Fl 33408

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: October 30, 1996

Description of amendment request: The proposed amendments will revise Technical Specification (TS) 3/4.9.10, ''Refueling Operations, Water Level-Reactor Vessel.'' The Limiting Condition for Operation (LCO) specified for the minimum allowed refueling water level is not altered, but the Applicability, Action, and Surveillance Requirements are changed to remove inconsistencies with the definition of Core Alterations, and to achieve consistency with the generic Standard Technical Specifications for Combustion Engineering Plants (NUREG-1432). An editorial change is proposed for TS 3/4.9.9, ''Refueling Operations, Containment Isolation System,'' and, for St. Lucie Unit 1, the LCO is modified to conform with other related refueling specifications.

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.

Certain evolutions performed with the UGS [upper guide structure] in place are not Core Alterations, and the revised LCO 3/4.9.10 will allow these activities to be performed at water levels other than prescribed by the existing LCO. Since these activities are performed with the UGS in place, the probability that a fuel handling accident would occur is not impacted by the proposed changes. The minimum water level required for Core Alterations and movement of irradiated fuel in containment is not altered by the proposed changes, nor are any assumptions or conditions changed that were used as inputs to the evaluation of fuel handling accident consequences. The changes to Specification 3/4.9.9 are administrative in nature and resolve an inconsistency between the operability requirements for the containment isolation system and the containment penetrations that the system would isolate at PSL1 [Plant St. Lucie 1]. Therefore, operation of either facility in accordance with its proposed amendment 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 changes are administrative in nature, in that the changes do not involve the addition or modification of equipment nor do they alter the design of plant systems. New failure modes are not introduced, and the physical plant or the modes of plant operation defined in the Facility License are not altered. Therefore, operation of either facility in accordance with its proposed amendment 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 safety margin associated with a fuel handling accident is determined, in part, by the minimum refueling water level allowed for conducting Core Alterations and movement of irradiated fuel in containment. The minimum water level required by LCO 3/4.9.10, or other factors considered as inputs to the safety analysis, is not changed by the proposed amendments. The revised applicability requirements for LCO 3/4.9.9 at PSL1 will allow the containment isolation system to be inoperable only during those Mode 6 conditions where Core Alterations or irradiated fuel movements within containment are not in progress, or each required containment penetration is otherwise closed. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on thisreview, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendmentrequest 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, 11770 US Highway 1, North Palm Beach, Fl 33408

NRC Project Director: Frederick J. Hebdon

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: October 23, 1996, as supplemented by letter dated November 6, 1996.

Description of amendment request: The proposed amendment would revise Technical Specification 3.4.6.1, regarding reactor coolant system leakage detection instrumentation, to adopt the requirements found in NUREG-1431, ''Standard Technical Specifications Westinghouse Plants,'' for the reactor coolant system leakage detection instrumentation.

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

The proposed change reduces the number of containment atmospheric radioactivity channels which must be OPERABLE when operating in MODES 1, 2, 3, and 4 from two to one. This change does not significantly increase the probability or consequences of a previously evaluated accident since the plant will continue to have diverse and independent means of detecting significant changes in the amount of leakage from the RCS [reactor coolant system]; the normal sump level and flow monitoring system, at least one of the two containment atmospheric radiation monitors, and the periodic precision RCS water inventory balance required by Technical Specification surveillance requirement 4.4.6.2.1.c. In addition, STP [South Texas Project] design includes advanced trending displays which can assist in detecting leakage based on changes in the volume control tank or pressurizer level. Other instruments, which are not listed in the Technical Specification related to leakage, but which can provide indication of leakage, are the containment pressure, temperature and humidity indicators. Good operating practice and commercial risk associated with long term inoperability of both monitors assures that an inoperable containment atmospheric radiation monitor will be promptly returned to service.

The proposed change also revises the limitation on continued operation with both containment atmospheric radiation monitors inoperable from 72 hours to 30 days. This change is based on the continued availability of diverse and redundant instrumentation discussed above to detect and indicate RCS leakage.

The Actions required as a result of this change include analysis of a containment atmospheric grab sample or performance of a precision RCS water inventory balance in accordance with surveillance requirement 4.4.6.2.1.c. The containment normal sump level flow monitoring system will also promptly identify changes in RCS leakage. Other installed instrumentation, such as containment pressure, temperature, and humidity, will provide indications of significant increases in leakage. Slower increases will be detected by the daily inventory balance or the daily grab samples analysis, and the three day inventory balance.

Inoperability of the on-line automatic containment normal sump level and flow monitoring system can be compensated for by the performance of a daily manual calculation, a precision RCS inventory balance as described in surveillance requirement 4.4.6.2.1.c, or the other available indications of increases in leakage such as the containment atmospheric radiation monitoring instruments and installed containment temperature, pressure and humidity instrumentation. The STP control room design also incorporates features which allow rapid detection of unexpected changes in the volume control tank and pressurizer level through available instrument trend displays. The combination of the compensatory measures, diverse and separate channels, and non-TS [non-technical specification] required instrumentation provides a sufficient level of detection to assure prompt identification and quantification of leakage with an inoperable containment normal sump level and flow monitoring system. The allowable outage time of 30 days provides assurance the normal containment sump level and flow monitoring system will be returned to service in a reasonable amount of time.

Based on the continued availability of adequate and redundant instrumentation to detect changes in RCS leakage rate, this 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.

The proposed change does not require the installation of any new or different kind of equipment. Nor does the change involve any significant new or different MODE of operation of the plant. The proposed change reduces the number of required containment atmospheric radiation monitors, and provides a 30 day allowed outage time for either the containment atmosphere radioactivity monitor or the containment normal sump level and flow monitoring system. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

In addition, as described above, the proposed change does not significantly reduce a margin of safety. Small changes in RCS leak rates are typically detected over a relatively long period of time. Diverse instrumentation continues to be available to plant operators which will assist in early detection of any change. The STP design provides additional non-Technical Specification human factors which assist in assuring any changes in leakage will be quickly detected.

The proposed change extends the amount of time that the containment atmospheric radiation monitors may be inoperable. The extension is based on the continued availability of equipment which provides a level of detection capability adequate to detect increases in RCS leakage and which continues to be diverse and independent. This protection is afforded by the continued OPERABILITY of the containment normal sump level and flow monitoring system, the daily performance of a precision RCS inventory balance as described by surveillance requirement 4.4.6.2.1.c or the daily analysis of containment atmospheric grab samples, and other instrumentation such as pressure, temperature and humidity indicators.

The combination of the compensatory measures, diverse and separate channels, and non-TS required instrumentation provides a sufficient level of detection to assure prompt identification and quantification of leakage with an inoperable containment normal sump level and flow monitoring system. Additionally, the compensatory measure of performing either a daily manual calculation or precision RCS inventory balance, provides assurance that the level of safety is maintained when the containment normal sump level and flow monitoring system is inoperable. The allowable outage time of 30 days provides assurance the normal containment sump level and flow monitoring system will be returned to service in a reasonable amount of time.

Based on the compensatory actions and available installed equipment, the proposed changes do not represent 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 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: William D. Beckner

Northern States Power Company, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota

Date of amendment requests: August 15, 1996

Description of amendment requests: The proposed amendments would revise the Containment Cooling Systems Limiting Conditions for Operation Technical Specifications to bring them into conformance with recently completed system analyses by no longer permitting both containment spray pumps to be inoperable at the same time.

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

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

Operation of the Prairie Island plant in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. None of the proposed changes involve a physical modification to the plant.

These changes will require operability of at least one containment spray pump at all times and reduces the spray additive tank allowable outage time from 72 hours to 24 hours. Both of these changes are more conservative and safer than currently required in the Prairie Island Technical Specifications. These proposed changes do allow one containment fan cooler train out of service for 7 days instead of 72 hours as allowed by current Technical Specifications. Recent plant analyses confirm that one containment fan cooler train with one containment spray train is sufficient to meet the system design bases. Since the probability of an accident occurring is low while one containment fan cooler train is out of service, the probability and consequences of an accident are not significantly increased.

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

2. The proposed amendment[s] will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes, in themselves, do not introduce a new mode of plant operation, surveillance requirement or involve a physical modification to the plant.

The proposed changes do require more restrictive, safer containment spray train operability. The proposed changes also allow one containment fan cooler train to be out of service for 7 days instead of 72 hours as allowed by the current Technical Specifications. However, this change does not create the possibility of a new kind of accident.

The proposed changes do no alter the design, function, or operation of any plant components and therefore, no new accident scenarios are created.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created by these amendments.

3. The proposed amendment[s] will not involve a significant reduction in the margin of safety. This License Amendment Request require[s] one containment spray train to be operable at all times which is more restrictive than current Technical Specifications and thus the margin of safety is not reduced.

This License Amendment Request will also allow one containment fan cooler train to be out of service for 7 days instead of 72 hours as allowed by the current Technical Specifications. Since the remaining containment cooling components can mitigate an accident and the probability of a design basis accident are low during this time, this change does not significantly reduce the plant margin of safety.

Therefore, a significant reduction in the margin of safety would not be involved with these amendments.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation [of] the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by Nuclear Regulatory Commission regulations in 10 CFR Part 50, Section 50.92.

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 requests involve no significant hazards consideration.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401

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

NRC Project Director: John N. Hannon

Northern States Power Company, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota

Date of amendment requests: September 24, 1996, as supplemented October 17, 1996.

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant to allow use of an alternate steam generator tube repair criteria (elevated F-star or EF*) in the tubesheet region when used with the repair method of additional roll expansion. The amendments incorporate revised acceptance criteria for tubes with degradation in the tubesheet region and enable the licensee to avoid unnecessary plugging and sleeving of steam generator tubes.

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

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

The supporting technical and safety evaluations of the subject criterion demonstrate that the presence of the tubesheet will enhance the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The results of hardrolling of the tube into the tubesheet is an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. The radial preload developed by the rolling process will secure a postulated separated tube end within the tubesheet during all plant conditions. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA [loss-of-coolant accident] loadings.

The EF* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the EF* distance, regardless of the extent of the tube degradation. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, tube rupture and pullout is not expected for tubes using the EF* criterion. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Prairie Island Plant USAR [updated safety analysis report]. For plants with partial depth roll expansion like Prairie Island, a postulated tube separation within the tube near the top of the roll expansion (with subsequent limited tube axial displacement) would not be expected to result in coolant release rates equal to those assumed in the USAR for a steam generator tube rupture event due to the limited gap between the tube and tubesheet. The proposed plugging criterion does not adversely impact any other previously evaluated design basis accident.

Leakage testing of roll expanded tubes indicates that for roll lengths approximately equal to the EF* distance, any postulated faulted condition primary to secondary leakage from EF* tubes would be insignificant.

2. The proposed amendment[s] will not create the possibility of a new or different kind of accident from any accident previously analyzed.

Implementation of the proposed EF* criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to initiate an accident outside of the region of the expanded portion of the tube. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity will be maintained. Tube bundle leaktightness will be maintained such that any postulated accident leakage from EF* tubes will be negligible with regard to offsite doses.

3. The proposed amendment[s] will not involve a significant reduction in the margin of safety.

The use of the EF* criterion has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Reg Guide 1.121 [≥Bases for Plugging Degraded PWR Steam Generator Tubes≥] (intended for indications in the free span of tubes) and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the EF* criterion is any degradation indication in the tubesheet region, more than the EF* distance below the bottom of the transition between the roll expansion and the unexpanded tube. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code used in steam generator design. The EF* distance has been verified by testing to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inner diameter based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the USAR accident analyses.

Implementation of the tubesheet plugging criterion will decrease the number of tubes which must be taken out of service with tube plugs or repaired with sleeves. Both plugs and sleeves reduce the RCS (reactor coolant system) flow margin; thus, implementation of the EF* criterion will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR Part 50, Section 50.92.

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 requests involve no significant hazards consideration.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401

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

NRC Project Director: John N. Hannon

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

Date of amendment request: June 10, 1996, as supplemented July 25, 1996

Description of amendment request: The proposed amendment would change the differential temperature Technical Specification Allowable Values and Trip Setpoints for the Reactor Water Cleanup penetration room steam leak detection function.

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 [of occurrence] [sic] or consequences of an accident evaluated.

FSAR section 5.2.5.1.3 addresses the ambient and differential room ventilation temperature leakage detection. This section states:

''...switch setpoints are based on the temperature rise resulting from a leak at system conditions corresponding to full reactor power.''

NRC Safety Evaluation on the RWCU system steam leak detection system (related to Amendment Number 123 to License NPF-14 and Amendment Number 90 to License NPF-22) reviewed and found acceptable the PP&L criteria for calculating the leak detection setpoints for the RWCU system, which include:

1. Setpoints are selected to detect and isolate a leak that is normally less than 25 gpm and below the flow rate corresponding for the critical crack size for the system piping.

2. Setpoints are set high enough to avoid inadvertent isolation caused by normal temperature transients or abnormal transients caused by non-leak conditions (such as loss of ventilation).

This NRC SER also stated that a leak rate of 25 gpm is less than those leak rates associated with the onset of unstable pipe ruptures. This fact is also shown in FSAR figure 5.2-10. This value of 25 gpm constitutes the design basis for the steam leak detection system.

The mixing and liquid energy addition assumption changes in the analysis do not affect this design basis. The analysis calculates the resulting room temperature increase from a 25 gpm leak. In fact, the new assumptions provide a more accurate yet conservative prediction of room temperature increases. Therefore, operation of the system is improved.

The proposed change leads to higher calculated room temperatures to be used in the differential temperature setpoint calculations. The engineering study was reviewed to determine if the higher calculated temperatures would have a negative impact on the High Energy Line Break and Leak Analysis environmental study which provides the basis for equipment qualification.

In determining the room temperatures, the engineering study considers ambient temperature setpoints at which the leaks will be isolated. The proposed action will not change the ambient temperature setpoints, and actuation of these instruments will ensure that the results of the engineering study will not be adversely affected. Therefore, no impact on equipment qualification is being introduced by this change.

FSAR chapter 15 was reviewed for potential impacts on the accident analyses. The 25 gpm leak outside containment is not specifically analyzed in FSAR chapter 15. However, other conditions which result in coolant leakage outside containment are analyzed in section 15.6.2 (Instrument Line Break) and 15.6.4 (Steam System Piping Break Outside Containment). As stated in the NRC SER, the 25 gpm RWCU leak rate is bounded by the analysis in FSAR section 15.6.4. FSAR section 15.6.2 also states that leak detection actuations will initiate operator actions, a fact that is not affected by the proposed change. Therefore, based on a review of FSAR chapter 15 it was concluded that no impact on the analyzed accident scenarios is created by the proposed change.

Based on the above discussions, it is demonstrated that the proposed change will not adversely impact system function or equipment. System performance will actually be improved since the new setpoints eliminate spurious isolations resulting from a less accurate model. The setpoint change has no impact on any equipment important to safety or any accidents previously analyzed in the FSAR. Therefore, the proposed change does not involve a significant increase in the probability of occurrence 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 action does not create the possibility of a new or different kind of accident from any accident previously evaluated. Neither the system design basis nor the system function will be adversely affected. System performance will be enhanced since spurious differential temperature actuations will be reduced as a result of using the more accurate, yet conservative, COTTAP model. In addition to this, redundant temperature isolation function will continue to be provided by the existing high ambient temperature detectors.

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

The proposed action does not involve a significant reduction in a margin of safety. The Technical Specification basis for the setpoints is to detect a leak below the flow rate corresponding to critical crack size for the system piping. As stated previously, the 25 gpm flow rate is an acceptable flow rate and is used to calculate the new temperatures.

Although the newly calculated RWCU penetration room temperatures will be higher (due to the improved model), the isolation actuation will be initiated by the high ambient temperature detectors before the penetration room temperatures reach the newly calculated values, as would happen under the old model. Therefore, system response is not adversely affected.

The current temperature values lead to differential temperature setpoints which are too low, causing spurious isolations. The use of the new temperature values will reduce the number of spurious isolations, reducing unnecessary challenges to safety systems during normal plant operations.

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

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

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

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: September 18, 1995

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Table 4.3.1.1-1, ''Reactor Protection System Instrumentation Surveillance Requirements'' to reflect the change in the calibration frequency for the Local Power Range Monitor (LPRM) signal from every 1000 Effective Full Power Hours (EFPH) to every 2000 Megawatt Days per Standard Ton (MWD/ST).

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

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

The change in the calibration frequency of the Local Power Range Monitor (LPRM) signal does not make any physical change to the fuel or the manner in which the fuel responds to a transient or accident. The proposed TS change does not affect the fundamental method by which the LPRMs are calibrated. Also, the LPRM calibration frequency is not considered an initiator of any events analyzed in the SAR. Therefore, calibrating the LPRMs on a different frequency will not increase the probability of occurrence of an accident previously evaluated in the SAR.

The resulting nodal power uncertainty does not exceed the nodal power uncertainty accounted for in the existing Minimum Critical Power Ratio (MCPR) Safety Limit; thus, the MCPR Safety Limit is not affected by this TS Change, and, therefore, the initial conditions of any accident are unchanged. Since the calibration frequency change will not affect the course of any evaluated accident, the consequences of an accident previously evaluated in the SAR will not be increased.

Therefore, the proposed TS change does 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 change in the calibration frequency of the Local Power Range Monitor (LPRM) signal does not make any physical change to the plant or the manner in which the equipment responds to a transient or accident. The proposed TS change does not introduce a new mode of plant operation and does not involve the installation of any new equipment or instrumentation. The fuel will continue to be operated to the same safety limits since the Minimum Critical Power Ratio (MCPR) Safety Limit remains unchanged due to this TS change.

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

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

The following TS Bases were reviewed for potential reduction in the margin of safety:

2.0 Safety Limits and Limiting Safety System Settings;

3/4.1 Reactivity Control Systems;

3/4.2.1 Average Planar Linear Heat Generation Rate;

3/4.2.3 Minimum Critical Power Ratio:

3/4.2.4 Linear Heat Generation Rate;

3/4.3.1 Reactor Protection System Instrumentation;

3/4.3.6 Control Rod Block Instrumentation;

3/4.3.7.7 Traversing In-Core Probe System;

The GE Thermal Analysis Basis (GETAB) determination of the Minimum Critical Power Ratio (MCPR) Safety Limit allows a maximum total nodal uncertainty of the Traversing In-Core Probe (TIP) readings of which the Local Power Range Monitor (LPRM).

Update uncertainty is a part. The change in LPRM calibration frequency results in an LPRM Update uncertainty which, when combined with the other uncertainties which comprise the total TIP readings uncertainty, yields a total TIP readings nodal power uncertainty of less than the allowed GETAB uncertainty. Thus the change in LPRM calibration frequency will not affect the MCPR Safety Limit.

The LPRMs are utilized as input to the Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) systems. The primary safety function of the APRM system is to initiate a scram during core-wide neutron flux transients before the actual core-wide neutron flux level exceeds the safety analysis design basis. This prevents fuel damage from single operator errors or equipment malfunctions. The APRMs are calibrated at least once per week to the plant heat balance, utilize a radially and axially diverse group of LPRMs as input and are utilized to detect changes in average, not local, power changes. Therefore, the effects of changing the LPRM calibration frequency on the APRM system responses will be minimal due to any individual LPRM drift being practically canceled out (due to diversity of input) and/or due to the frequent recalibration of the APRMs to an independent power calculation (the heat balance). Thus, changing the LPRM calibration frequency will not impact the capability of the APRM system to perform the scram function, and there is no impact on transient delta-CPRs.

The RBM system is utilized in the mitigation of a Rod Withdrawal Error (RWE) event. The RBM system is designed to prevent the operator from increasing the local power significantly when withdrawing a control rod. Under Average Power Range Monitor - Rod Block Monitor Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) on each selection of a control rod, the average of the assigned, unbypassed LPRMs is adjusted to equal a 100% reference signal for each of the two RBM channels. Each RBM channel automatically limits the local thermal margin changes by limiting the allowable change in local average neutron flux to the RBM setpoint. If the local average neutron flux change is greater than that allowed by the RBM setpoint, within either RBM channel, the rod withdrawal permissive is removed preventing further rod movement. Since the change in local neutron flux is calculated from the change in the average of the LPRM readings, and calibrated on every rod selection to the reference signal, offsets in individual LPRM readings due to calibration differences are effectively eliminated for a given RBM setpoint. Therefore, the constraints on the withdrawal of any given rod are unchanged, and there will not be any increase in RWE delta-CPR.

Since the MCPR Safety Limit is unaffected and the delta-CPR values are unchanged, the cycle CPR Operating limits are unchanged due to this TS change. Therefore, the proposed change in the frequency of LPRM signal calibration does not result in 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 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: Pottstown Public Library, 500 High Street, Pottstown, PA 19464

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

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: May 3, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Surveillance Requirements 4.6.5.3.a and 4.6.5.4.a to modify specific requirements to perform surveillance flow testing of the Standby Gas Treatment and Reactor Enclosure Recirculation Systems from monthly to quarterly.

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

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

The proposed TS changes do not involve any physical changes to plant systems or equipment. The proposed TS changes only change the Surveillance Requirements (SRs) surveillance test frequency pertaining to flow testing of the SGTS and RERS from monthly to quarterly. The periodic surveillance test frequencies provide adequate assurance that the equipment tested will remain in an operable condition. The test frequency interval for the flow testing of the SGTS and RERS was determined from the regulatory position in USNRC Regulatory Guide 1.52, ''Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Clean-up System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants''. As stated in Regulatory Position C.4.d, ''... each Engineered Safety Feature (ESF) atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture on the absorbers and HEPA filters.''

System operation on a monthly basis for the purpose of preventing moisture buildup on the absorbers as described in R.G. 1.52 is not required at Limerick due to the continuous dry instrument air purge described previously in the Safety Assessment section of this submittal. Therefore a change in the interval between tests from monthly to quarterly will not result in moisture accumulation which would reduce the capability of the absorber to remove the iodine species from the exhaust air flow stream.

The SGTS components are common to both units and must be run with the associated RERS for the surveillance test for each unit. The currently specified test frequency results in the SGTS being run at least twice per month or as many as eight (8) times per quarter for this surveillance, in addition to other required system surveillance tests which require the use of the components in this system. A change in surveillance test frequency from monthly to quarterly would reduce the wear on system components and thereby reduce the associated system downtime for maintenance and repairs. The consequent increased availability provides greater assurance that the system will be able to perform its mitigation function following any postulated accident.

Surveillance test frequency on a quarterly interval is considered adequate to verify operability, as demonstrated by the required quarterly test interval for other equipment important to safety which have a similar function, such as the requirement for quarterly verification of the isolation time of the secondary containment and refueling area isolation valves, as required by LGS TS Sections 4.6.5.2.1 and 4.6.5.2.2.

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

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

The proposed TS changes only involve changes to the frequency in which the specified surveillances tests are performed. The proposed TS changes do not physically change the design or intended function of the systems, structures, or components associated with the SGTS or RERS. There will be no change to the existing redundancy of systems and components. The proposed change in surveillance test frequency will not introduce the possibility of any failure mechanisms of a different type than those already evaluated in the SAR. The existing components will not be used in any different manner and no new components will be added. Therefore with no physical changes and no new or different manner of system operation, no new failure mechanisms or equipment failure modes are created.

Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any 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 LGS TS Bases has not been reduced. The specific basis for the 31 day surveillance interval is not given in the LGS TS Bases section nor in the LGS UFSAR Sections 6.5.1 or 9.4.2 which discuss the subject systems. However, Regulatory Position C.4.d of Regulatory Guide 1.52, Revision 2, relating to maintenance requirements, recommends:

≥Each ESF atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture on the absorbers and HEPA filters.''

The Bases for Surveillance Requirements (SR) 3.6.4.3.1 in the Standard Technical Specifications for General Electric Plants, BWR/4, which corresponds to the subject LGS TS test, also notes the need for ten (10) hours of operation per month for elimination of moisture in the filters.

The basis for the requirement for a monthly test with the heaters energized is clearly related to the desired elimination of moisture in the filters and absorbers. However, LGS UFSAR Table 6.5-2 states that LGS does not conform to R.G. 1.52, Position C.4.d because the SGTS and RERS trains are ''continuously purged with dry instrumentation air to prevent build-up of moisture.'' UFSAR Sections 6.5.1.1.2 and 6.5.1.3.2 provide additional discussion of this method of moisture control.

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

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

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

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

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: September 27, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would increase the Reactor Enclosure Secondary Containment maximum inleakage rate. This change will also impact secondary containment drawdown time and system flow rate assumptions, thereby, affecting charcoal filter bed efficiency and post accident dose analysis.

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

Changing the Reactor Enclosure post drawdown inleakage rate from 1250 cfm to 2500 cfm does not involve any changes to the function or operation of any plant component or safety related system. The Reactor Enclosure Recirculation System (RERS) and the Standby Gas Treatment System (SGTS) will maintain their design function by mitigating the radiological consequences of the analyzed accident and mitigating the post LOCA temperatures within the Reactor Enclosures. No analyzed accident initiating events are impacted, no new accident initiators are created, and no new failure modes are created. There are no changes to the redundancy, separation, quality assurance or fire protection requirements for the associated components and systems.

The proposed changes to the LGS adsorber bed residence time will no longer fully meet the literal design guidance provided in Regulatory Guide (RG) 1.52, ''Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filter and Adsorption Units of Light-Water-Cooled Nuclear Power Plants,'' Revision 2, March 1978. This is because LGS's unique, yet more conservative, adsorber bed design is not addressed by the RG residence time design guidance. However, the LGS SGTS charcoal adsorbers still conform to the design function described in RG 1.52, based on the following: The LGS design with increased inleakage will continue to conform to the three conditions specified by RG 1.52, Position C.6.a, in order to maintain an assigned decontamination efficiency of 99%; there is a conservative amount of charcoal adsorber material provided by the LGS design, based on calculations performed in accordance with RG 1.3 ''Assumptions Used For Evaluating The Potential Radiological Consequences of a Loss of Coolant Accident For Boiling Water Reactors; and the LGS charcoal bed design is more conservative than the RG 1.52 design guidance, based on data (i.e., Iodine Penetration vs. Air Velocity) published by the charcoal manufacturer.

Therefore, the probability of occurrence and the consequences of a malfunction of equipment important to safety is not increased. Also, the probability of occurrence of an accident previously evaluated is not increased. However, the proposed changes do affect the leak tightness of the Unit 1 and Unit 2 Reactor Enclosure, which increases the consequences of a postulated accident previously evaluated.

Changing the Reactor Enclosure post drawdown inleakage rate from 1250 cfm to 2500 cfm will result in an increase in the calculated LOCA/LOOP Design Basis Accident (DBA) off-site and on-site doses. 10 CFR Part 100, and 10 CFR Part 50 Appendix A, General Design Criteria (GDC) 19, establish reference dose values used to determine site suitability and provide reasonable assurance that the facility can be operated following the analyzed accident without undue risk to the health and safety of the public. The proposed TS changes will increase the SGTS drawdown time from 2 minutes and 20 seconds to 15 minutes and 30 seconds. The drawdown time increase will not prevent the RERS/SGTS from performing all of their safety related functions. However, because it is conservatively assumed that all radioactive material released during the drawdown period is unfiltered, and because the drawdown period has been extended whereby more unfiltered radioactive material is assumed to be released following the DBA, there is a corresponding increase in the calculated Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room doses. It is also assumed that the SGTS exhausts at the maximum inleakage rate throughout the entire DBA evaluation period (i.e., 30 days) where an increase in the maximum inleakage rate would also contribute to higher postulated EAB, LPZ, and Control Room doses. However, the proposed calculated doses do not exceed 10 CFR Part 100, or 10 CFR Part 50, Appendix A, DGC 19 reference doses.

Since the proposed doses resulting from the changes remain below 10 CFR Part 100, and 10 CFR Part 50, Appendix A, these proposed changes will not significantly increase the consequences of an accident previously evaluated.

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

Changing the Reactor Enclosure post drawdown inleakage rate from 1250 cfm to 2500 cfm is not an accident initiator nor does it result in the occurrence of an accident. The changes do not affect the function or operation of any plant component or safety related system nor do they create any new failure modes.

In addition, the proposed changes do not involve any changes to the function or operation of any plant system or component nor will they adversely affect the Reactor Enclosure post LOCA environmental conditions. Furthermore, these changes will not create any new or different failure modes for the equipment important to safety within the Reactor Enclosure Secondary Containment.

Therefore, the possibility of an accident of a different type or a different type of malfunction of equipment important to safety than previously evaluated is not created.

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

Changing the Reactor Enclosure post drawdown inleakage rate from 1250 cfm to 2500 cfm will result in reducing the margin of safety as defined in the LGS Updated Final Safety Analysis Report (UFSAR) relative to the off-site and on-site doses following a LOCA/LOOP DBA, and an increase of the UFSAR specified system drawdown time. From a system perspective, increasing the SGTS drawdown time from 2 minutes and 20 seconds to 15 minutes and 30 seconds will not prevent the RERS/SGTS from performing all of their safety related functions. There will be a postulated increase in the corresponding EAB, LPZ, and Control Room doses, since it is assumed that fuel damage occurs coincident with the LGS DBA (i.e, at time = 0), all radioactive material released during the drawdown time is unfiltered, and the drawdown time is proposed to be extended whereby more unfiltered radioactive material could be released. It is also assumed that the SGTS exhausts at the maximum inleakage rate throughout the entire DBA evaluation period (i.e., 30 days) where an increase in the maximum inleakage rate would also contribute to higher postulated EAB, LPZ, and Control Room doses. However, these calculated doses will remain below 10 CFR Part 100, and 10 CFR Part 50, Appendix A, GDC 19 reference doses.

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

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: October 1, 1996

Description of amendment request: The proposed amendment would allow for a one-time extension of the surveillance intervals for the containment isolation valve (CIV) seat leakage test, the isolation valve seal water test, the boron injection tank leakage test, the containment spray nozzle test, and the city water backup to the auxiliary boiler feed pump test. These tests would be performed during the refueling outage scheduled to begin in April 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:

Regarding the Containment Isolation Valve seat leakage and Isolation Valve Seal Water tests:

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

Response:

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The probability of a previously evaluated accident will not increase because CIV leakage does not provide any role in accident initiation. The CIVs provide containment isolation following a design basis accident.

The consequences of an accident previously evaluated will not significantly increase because the CIV leakage measurements contain significant margin to a more restrictive criteria based on the requested surveillance interval extension. As discussed in Section II, ''Evaluation of Changes,'' [see application dated October 1, 1996] based on an evaluation of past CIV leak tests, the proposed change will not result in an increase in containment leakage because the measured leakage in previous CIV leak tests shows large margin to a more restrictive criteria based on the requested surveillance interval extension. Also, the latest test of IVSWS [isolation valve seal water system] satisfied the established acceptance criteria.

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

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change only provides for a relatively short, one-time extension of the current leak-test interval for certain containment isolation valves. The proposed change does not involve the addition of any new or different type of equipment, nor does it involve operating equipment required for safe operation of the facility in a manner different from that addressed in the Final Safety Analysis Report. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response:

The proposed amendment does not involve a significant reduction in a margin of safety. The proposed change, for a one-time extension of the test interval, will not result in a significant reduction in a margin of safety because the test interval is being extended by only a short period and the measured leakage in previous CIV leak tests shows large margin to a more restrictive criteria based on the surveillance interval extension. In addition, the online leakage monitoring capability of the WCCPPS [weld channel containment penetration pressurization system] helps ensure that changes in CIV leakage during the extension period will be detected. Therefore, this change does not create a significant reduction in a margin of safety.

Regarding the Boron Injection Tank (BIT) leakage test:

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

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change will not degrade the integrity of the BIT piping outside containment because no time dependent failure trends were observed in the review of past test results. The probability of a previously evaluated accident will not be increased because BIT leakage does not provide any role in accident prevention. The BIT leakage test only verifies that the BIT and associated piping meet specified leakage limits.

The consequences of an accident previously evaluated will not significantly increase because the BIT leakage test results show large margins to the allowable leakage limit.

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

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change does no[t] involve the addition of any new or different type of equipment, nor does it involve operating equipment required for safe operation of the facility in a manner that's different from that addressed in the Final Safety Analysis Report. Also, the increased surveillance interval (one-time only) will not adversely affect the integrity of the BIT piping and will not result in any new failure modes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response:

The proposed license amendment does not involve a significant reduction in a margin of safety. Because of the large margin between the previous test and the allowable leak rate limit, the proposed change, for a one-time extension of the test interval, for the BIT leakage test does not adversely affect the performance of any safety related system, component, and does not result in increased severity of any of the accidents considered in the Final Safety Analysis Report. Based on past test results, the one-time extension of the leak test interval does not involve a significant reduction in a margin of safety.

Regarding the Containment Spray Nozzle test:

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

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. As discussed in Section II, ''Evaluation of Changes,'' [see application dated October 1, 1996] based on an evaluation of past test results the proposed change will not degrade the reliability of the Containment Spray Nozzles because no time dependent failure trends were observed in the data review. The probability of a previously evaluated accident will not be increased because the Containment Spray Nozzles do not provide any role in accident prevention. The Containment Spray Nozzles provide a uniform spray distribution for containment cooling following postulated post-accident conditions.

The consequences of an accident previously evaluated will not increase because the Containment Spray Nozzle reliability is not degraded by this change.

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

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change does not involve the addition of any new or different type of equipment, nor does it involve operating equipment required for safe operation of the facility in a manner that is different from that addressed in the Final Safety Analysis Report. Also, the increased surveillance interval (one-time only) w[i]ll not adversely affect the functioning of the Containment Spray Nozzles and will not result in any new failure modes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response:

The proposed license amendment does not involve a significant reduction in a margin of safety. The proposed change, for a one-time extension of the test interval, for the Containment Spray Nozzles does not adversely affect the performance of any safety related system, component, or instrument, or safety system setpoints and does not result in increased severity of any of the accidents considered in the Final Safety Analysis Report. Based on past test results, the one-time extension of the functional test interval will not adversely affect the functioning of the Containment Spray Nozzles. Therefore, this change does not create a significant reduction in a margin of safety.

Regarding the City Water Backup test:

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

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change will not degrade the reliability of the City Water Backup Supply Valves for the AFW [auxiliary feedwater] System because no time dependent failure trends were observed in the review of past test results. The probability of a previously evaluated accident will not increase because the City Water Backup Supply Valves for the AFW System do not provide any role in accident prevention. The City Water Backup Supply Valves for the AFW System only provides a diverse source of water for the AFW system.

The consequences of an accident previously evaluated will not significantly increase because the City Water Backup Supply Valves for the AFW System are not assumed to function to mitigate any analyzed accident.

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

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change does not involve the addition of any new or different type of equipment, nor does it involve operating equipment required for safe operation of the facility in a manner that is different from that addressed in the Final Safety Analysis Report. Also, the increased surveillance interval (one-time only) will not adversely affect the functioning of the City Water Backup Supply Valves for the ABFP [auxiliary boiler feedpump] and will not result in any new failure modes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response:

The proposed amendment does not involve a significant reduction in a margin of safety. The proposed change, for a one-time extension of the test interval, for the City Water Backup Supply Valves for the ABFP does not adversely affect the performance of any safety related system, component, or instrument, or safety system setpoints and does not result in increased severity of any of the accidents considered in the Final Safety Analysis Report. Based on past test results, the one-time extension of the functional test interval will not adversely affect the functioning of the City Water Backup Supply Valves for the AFW System. Therefore, this change does not create 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: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Acting

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

Date of amendment request: October 25, 1996

Description of amendment request: The proposed change to Hope Creek Technical Specification (TS) 3/4.1.3.5, ''Control Rod Scram Accumulator'', would: 1) permit a separate entry into a Technical Specification action statement for each inoperable control rod; 2) provide more specific applicability for required actions in operational condition 1 or 2 with one inoperable control rod scram accumulator (reactor pressure of greater than or equal to 900 psig would be specified); 3) provide more specific actions (verify charging water pressure) for two or more inoperable control rod scram accumulators and reactor pressure is greater than or equal to 900 psig; 4) provide more specific actions when reactor pressure is less than 900 psig and one or more control rod scram accumulators are inoperable (verify insertion of control rods associated with inoperable accumulators and verify that charging water header pressure is greater than or equal to 940 psig); and 5) provide specific actions in operational condition 5 with one or more withdrawn control rods inoperable; and 6) eliminate the requirements to perform a 18-month channel functional test of the leak detectors and the 18-month channel calibration of the pressure detectors.

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 change incorporates the appropriate content of the improved BWR/4 Standard Technical Specifications, NUREG-1433, for Control Rod Scram Accumulators.

The proposed Technical Specification and required Action completion times are consistent with or more conservative than those approved for use in the improved Technical Specifications for inoperable control rod scram accumulators. In addition, the proposed surveillance requirements for the control rod scram accumulators are sufficient to adequately demonstrate operability as stated in the Bases for the improved Technical Specifications. Further, the proposed changes enhance the current Hope Creek Technical Specifications by reflecting improved techniques collectively learned by the industry. Therefore, the proposed changes do not significantly increase the risk or consequences of any accidents previously evaluated.

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

Neither the mechanism for initiating or completing a scram is modified by this proposed change. There are no physical changes to plant equipment proposed in the application. The proposed change does not create a means by which the scram function could be impeded or prevented. The proposed change is functionally equivalent to the current Technical Specifications, but provides additional operational flexibility to diagnose and resolve equipment issues that do not impact operability of the control rods before taking proscriptive actions which result in significant plant transients (i.e. full power scram).

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

The operability of the accumulators and the scram function of the control rod drive system protects the Safety Limit Minimum Critical Power Ratio as well as the 1% cladding plastic strain fuel design limit. The proposed change does not reduce a margin of safety as defined in the Bases of the Technical Specification since the proposed change does not affect the maximum allowable scram times for control rods, nor does it change the maximum allowable number or minimum separation of inoperable control rods. The proposed change does not modify any instrument setpoints or functions. The proposed change will either maintain the present margins of safety or increase them, by reducing the need for unnecessary challenges to the reactor protection system and resulting plant shutdowns, while still maintaining the capability to complete a reactor scram.

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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: October 29, 1996

Description of amendment request: The proposed amendment would revise the mode of applicability for the motor-driven auxiliary feedwater (AFW) pump actuation on opening of the main feedwater (MFW) pump breakers to correct an error introduced during Amendment No. 61.

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

The less restrictive changes discussed in Section C.1 [of the licensee's application] do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes only correct an error which was introduced in Amendment No. 61 to the Ginna Station technical specifications. The changes revert the mode of applicability for the motor-driven AFW pump actuation on the opening of the MFW pump breakers to what existed previously. The change is essentially correction of a typographical error that was caused through use of the electronic version of NUREG-1431 in preparation of the Ginna Station ITS [Improved Technical Specifications]. There have been no subsequent plant modifications or changes to the accident analysis which would invalidate the previous NRC acceptance of only requiring this Function above 5% power. The accident analyses do not credit automatic initiation of AFW on MFW pump trip in MODE 2. As such, these changes do not impact initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation which existed prior to Amendment No. 61. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because there have been no subsequent plant modifications or changes to the accident analysis which would invalidate the previous NRC acceptance of only requiring this Function above 5% power. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005

NRC Project Director: S. Singh Bajwa, Acting

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: October 29, 1996

Description of amendment request: The proposed amendment would revise the Required Actions for the auxiliary feedwater (AFW) pump actuation on Steam Generator Level (SG) - Low Low logic to be consistent with those specified in NUREG-1431.

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

The less restrictive changes discussed in Section C.1 [of the licensee's application] do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes with respect to the Required Actions for AFW actuation on SG Level - Low Low logic provide consistency with NUREG-1431 by requiring an inoperable channel to be placed in the tripped condition within 6 hours. The affected logic then requires 1 of 2 channels in order to actuate such that there is no impact on any initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because the AFW actuation on SG Level - Low Low still remains capable of performing its function with an inoperable channel placed in the tripped configuration. These changes are also consistent with those provided in NUREG-1431. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005

NRC Project Director: S. Singh Bajwa, Acting

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: September 4, 1996

Description of amendment request: The proposed amendment to the Technical Specifications would allow the use of four lead test assemblies (advanced zirconium-based alloys) in the North Anna, Units 1 and 2, reactor cores.

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:

Operation of the four FCF [Framatome Cogema Fuels] lead test assemblies will not:

1.Involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The FCF lead test assemblies are very similar in design to the Westinghouse fuel that comprises the remainder of the core. The reload core design for North Anna cycles which incorporate the lead test assemblies will meet all applicable design criteria. In addition, the performance of the ECCS [emergency core cooling system] at North Anna Units 1 and 2 will not be affected by the insertion of the four lead test assemblies, so the criteria of 10 CFR 50.46 will be satisfied for use of these assemblies with fuel rods, guide thimble tubes, and instrumentation tubes fabricated with advanced zirconium-based alloys. The use of these fuel assemblies will not result in a change to the North Anna Units 1 and 2 reload design and safety analysis limits. The existing safety analyses based on the resident Westinghouse fuel will remain applicable for cycles which incorporate the lead test assemblies. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

2. Create the possibility for a new or different type of accident from any accident previously evaluated. The FCF lead test assemblies are very similar in design (both mechanical and composition of materials) to the resident Westinghouse fuel. North Anna cores which incorporate the lead test assemblies will be designed to meet all applicable design criteria and ensure that all pertinent licensing basis criteria are met. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. North Anna safety analyses based on the resident Westinghouse fuel will remain applicable for cores containing the lead test assemblies. All design and performance criteria will continue to be met and no single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alteration to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. Involve a significant reduction in the margin of safety. The use of the FCF lead test assemblies does not change the performance requirements on any system or component such that any design criteria will be exceeded, and will not cause the core to operate in excess of pertinent design basis operating limits. North Anna reload core designs for cycles which incorporate the lead test assemblies will specifically evaluate any pertinent differences between the lead test assemblies and the resident fuel, and will take into consideration the normal core operating conditions allowed in the Technical Specifications. Safety analyses based on the resident Westinghouse fuel will remain applicable for cores incorporating the FCF lead test assemblies. Analyses or evaluations will be performed each cycle to confirm that the criteria in 10 CFR 50.46 will be met. Therefore, the margin of safety as defined in the Bases to the North Anna Units 1 and 2 Technical Specifications is not significantly reduced.

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 Acting Project Director: Mark Reinhart

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: November 6, 1996

Description of amendment request: The proposed changes will modify the requirements for isolated loop startup to permit filling of a drained isolated loop via backfill from the reactor coolant system through partially open stop loop 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 changes will not:

1. Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The probability of occurrence of a positive reactivity addition accident is not being increased by the proposed Technical Specification change. The proposed restrictions on boron concentration and mixing, reactor coolant system inventory and reactivity and count rate monitoring provide a level of protection against reactivity addition accidents which is equivalent to that currently in place.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not introduce any new or unique failure modes or accident precursors. Eliminating the operability requirements for the loop stop valve interlocks does not create any new or different kind of accident scenario. Loop startup accidents in the various modes of operation have been analyzed. Operation of the loop stop valves will not change. New requirements have been imposed for the case of backfilling a drained loop from the reactor coolant system to ensure that core cooling and reactivity control are preserved throughout the backfill evolution.

3. Involve a significant reduction in any margin of safety. The new Technical Specification loop isolation and startup requirements for temperature, boron concentration, and shutdown margin fulfill the function of the loop stop valve interlocks. Therefore, the margin of safety as defined in any Technical Specification bases is not reduced.

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: Mark Reinhart (Acting)

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

Date of amendment request: October 31, 1996

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) by deleting the requirement for an annual submittal of a description of changes made pursuant to 10 CFR 50.59. Consistent with 10 CFR 50.59(b)(2), a description of changes will subsequently be included with the KNPP Updated Safety Analysis Report (USAR) update in accordance with 10 CFR 50.71(e). Additionally, the proposed amendment would correct minor administrative inconsistencies in the TS Table of Contents and in a footnote reference.

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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c)The NRC staff's review is presented below:

On August 31, 1992 (57 FR 39353), the NRC amended 10 CFR 50.59(b)(2) to reduce the regulatory burden on nuclear licensees. This action revised the requirements for the annual submission of reports for facility changes under 10 CFR 50.59. This action did not affect the substance of the evaluation or the documentation required for 10 CFR 50.59 type changes. It only affected the interval for submission of the information to the NRC. Instead of submitting the information annually, the information can be submitted on a refueling cycle basis, provided the interval between successive reports does not exceed 24 months.

In order to take advantage of this reduction in regulatory burden, the licensee has proposed an amendment to remove the submittal of a report of facility changes under 10 CFR 50.59 from the Technical Specification list of annual reporting requirements. Additionally, the licensee has proposed corrections to minor administrative inconsistencies in the TS Table of Contents and in a footnote reference. The proposed changes are administrative only and do 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.

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 Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497

NRC Project Director: Gail H. Marcus

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.

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

Date of application for amendment: May 1, 1996, as supplemented August 12, 1996.

Brief description of amendment: The amendment approves relocation of the administrative controls related to the quality assurance review and audit requirements of Section 6, Technical Specifications 6.5.B.8, ''Nuclear Safety Review and Audit Committee-Audits,'' from the Pilgrim Station Technical Specifications to the Boston Edison Quality Assurance Manual (BEQAM). This change is in accordance with the guidance contained in NRC Administrative Letter 95-06, ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance.'' In addition, the Safety Evaluation includes the NRC staff review and approval of the BEQAM changes in support of this amendment.

Date of issuance: November 12, 1996

Effective date: November 12, 1996

Amendment No.: 168

Facility Operating License No. DPR-35: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28605) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

Date of application for amendments: August 29, 1996, as supplemented on September 20, 1996, and October 4, 1996.

Brief description of amendments: The amendments change the Technical Specifications to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program,'' with an exception as detailed in the licensee's application.

Date of issuance: November 12, 1996

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

Amendment Nos.: 175 and 162

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

Date of initial notice in Federal Register: October 9, 1996 (61 FR 52964). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 12, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: August 14, 1996, as supplemented October 18, 1996, and related application of January 18, 1996

Brief description of amendment: The amendment revises the technical specifications (TS) to allow one-cycle deferral of the inspection of reactor coolant pump (RCP) flywheels.

Date of issuance: November 7, 1996

Effective date: November 7, 1996

Amendment No.: 175

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

Date of initial notice in Federal Register: September 24, 1996 (61 FR 50054) The October 18, 1996, letter provided an updated TS page. This change was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 7, 1996.No significant hazards consideration comments received: No.

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

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: December 14, 1994, as supplemented by letters dated May 16 and August 29, 1996

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

Date of issuance: November 12, 1996

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

Amendment Nos.: 170 and 152

Facility Operating License Nos. NPF-9 and NPF-17: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28612) The August 29, 1996, letter provided clarifying information that did not change the scope of the December 14, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 12, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

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

Date of amendment request: August 1, 1996

Brief description of amendment: The amendment revises the technical specifications to incorporate requirements for limiting the time that the hydrogen mixing isolation valves on the drywell are open. The amendment also changes the time from 7 days to 31 days to determine the cumulative time the valves are open.

Date of issuance: November 12, 1996

Effective date: November 12, 1996

Amendment No.: 89

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

Date of initial notice in Federal Register: September 25, 1996 (61 FR 50343) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 1996.No significant hazards consideration comments received. No.

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

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

Date of application for amendment: May 9, 1996, as supplemented by letter dated August 27, 1996.

Brief description of amendment: The amendment changed Surveillance Requirements (SRs) 3.4.4.3, Safety/Relief Valves, 3.5.1.7, Automatic Depressurization System Valves, and 3.6.1.6.1, Low-Low Set Valves, of the Technical Specifications and allows the licensee to perform the surveillance of the relief mode of operation of the safety/relief valves on the main steam lines without physically lifting the disk of a valve off the seat at power. The changes stated that the required operation of the valve to verify is that the relief-mode actuator strokes when the valve is manually actuated and the frequency of the surveillances are in accordance with the inservice testing program for the valves.

Date of issuance: November 18, 1996

Effective date: November 18, 1996

Amendment No: 130

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

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47971) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 1996. No significant hazards consideration comments received: No

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

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: June 3, 1996, as supplemented October 23, 1996

Brief description of amendment: The amendment clarifies a restriction on shutdown margin monitor operability while changing operational modes, so that it only limits reactivity changes caused by boron dilution and rod withdrawal. The amendment also corrects a technical specification numerical reference so that the specification number cited is in agreement with Amendment 99, dated December 29, 1994.

Date of issuance: November 14, 1996

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

Amendment No.: 131

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

Date of initial notice in Federal Register: June 20, 1996 (61 FR 31559) The October 23, 1996, letter provided clarifying information that did not change the scope of the June 3, 1996, application and the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 1996No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 30, 1996

Brief description of amendment: The proposed change to the anticipated transient without scram recirculation pump trip logic for the James A. Fitzpatrick Nuclear Power Plant allows for a high pressure trip setpoint which is dependent upon the number of safety/relief valves which are out of service.

Date of issuance: November 7, 1996

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

Amendment No.: 237

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34896) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 7, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: May 30, 1996, as supplemented October 17, and November 8, 1996

Brief description of amendment: The proposed amendment changes the FitzPatrick safety limit minimum critical power ratio from its current value of 1.07 for two recirculation loop operation to 1.09 and from 1.08 to 1.10 for single recirculation loop operation for the Cycle 13 operation.

Date of issuance: November 14, 1996

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

Amendment No.: 238

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34896) The October 17 and November 8, 1996 letters provided supplemental information that did not change the initial no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 1996.No significant hazards consideration comments received: No

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

Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: August 23, 1996, as supplemented by letters dated September 16, November 6, 11 and 14, 1996

Brief description of amendment: The amendment changes the Technical Specifications (TS) to allow installation of laser welded elevated tubesheet sleeves. Specifically, the amendment is for one cycle only for Farley Unit 2. Permanent, generic TS changes for Westinghouse laser welded sleeves for both units will be submitted prior to the next Unit 1 refueling outage currently scheduled for spring 1997.

Date of issuance: November 20, 1996

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

Amendment No.: 117

Facility Operating License No. NPF-8: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47982) The September 16, November 6, 11 and 14, 1996, letters provided clarifying information that did not change the scope of the August 23, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 20, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 17, 1995.

Brief description of amendments: These amendments revise the frequency of surveillance requirements for certain plant protective system instrumentation contained in Technical Specifications (TS) 3.3.1, ''Reactor Protective System (RPS) Instrumentation - Operating,'' TS 3.3.2, ''Reactor Protective System (RPS) Instrumentation - Shutdown,'' TS 3.3.3, ''Control Element Assembly Calculators (CEACs),'' TS 3.3.4, ''Reactor Protective System (RPS) Logic and Trip Initiation,'' TS 3.3.5, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' and TS 3.3.6, ''Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip.''

Date of issuance: November 18, 1996

Effective date: November 18, 1996, to be implemented within 30 days of the date of issuance.

Amendment Nos.: Unit 2 - 133 ; Unit 3 - 122

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

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45185) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 1996.No significant hazards consideration comments received: No.Temporary

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

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

Date of application for amendment: September 4, 1996

Brief description of amendment: This amendment revises Technical

Specification (TS) 6.2.3, ''Facility Staff Overtime,'' by removing specific overtime limits and working hours and by adding procedural controls to perform a monthly review of overtime hours.

Date of issuance: November 8, 1996

Effective date: November 8, 1996, to be implemented not later than 90 days after issuance

Amendment No.: 212

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

Date of initial notice in Federal Register: October 9, 1996 (61 FR 52970) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 8, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 18, 1996

Brief description of amendment: The amendment adopts ASTM D-3803-1989 as the laboratory testing standard for charcoal samples from the charcoal adsorbers in the auxiliary/fuel building emergency exhaust system.

Date of issuance: November 13, 1996

Effective date: November 13, 1996, to be implemented within 30 days of the date of issuance.

Amendment No.: 118

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

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42285) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 1996.No significant hazards consideration comments received: No.

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

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

Dated at Rockville, Maryland, this 26th day of November 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

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

[Doc. 96–30714 Filed 12–3–96; 8:45 am]

BILLING CODE 7590–01–F

NUCLEAR WASTE TECHNICAL REVIEW BOARD

Pahrump, Nevada: Yucca Mountain Testing and Exploration Program, Environmental Impact Statement, Interim Storage Studies, Transportation Infrastructure, Options for Reducing Hydrogeologic Uncertainties in the Proposed Repository Waste Emplacement Area, and Performance Assessment Issues; Board Meeting

Pursuant to its authority under section 5051 of Public Law 100–203, the Nuclear Waste Policy Amendments Act of 1987, the Nuclear Waste Technical Review Board will hold its winter meeting on Tuesday and Wednesday, January 28–29, 1997, in Pahrump, Nevada. The meeting will be held at the Bob Ruud Community Center, 150 N. Highway 160, Pahrump, Nevada 89048; Tel (702) 727–9991. Sleeping accommodations are available in the Saddle West Hotel, 1220 S. Highway 160, Pahrump, Nevada 89048; Tel (702) 727–1111; Fax (702) 727–5315. To receive the preferred rate, reservations must be made by December 27, 1996. The meeting is open to the public and will begin at 8:30 a.m. both days.

On the first morning, the Board will hear presentations by representatives of the Department of Energy (DOE) and its contractors on the exploration and testing program at Yucca Mountain, Nevada; plans for preparing the environmental impact statement for repository development; and generic studies on the development of an interim spent fuel storage facility. The Board is particularly interested in hearing about how long it would take to construct such a facility and to develop a transportation infrastructure to move significant quantities of waste. The Board plans to invite Nye County representatives to briefly summarize the results of their scientific investigations at the Yucca Mountain site.

The afternoon session will examine the issues associated with DOE plans to reduce, by late 1998, the current uncertainties about the movement of moisture through the proposed repository waste emplacement area. The focus will be on options for gathering additional data, including what data would be sought, and how the data would be obtained.

On the second day of the meeting, the morning session will address the transportation of waste to a potential repository at the Yucca Mountain site, including an update on the DOE's recent privatization initiative and on more local issues such as route selection and emergency preparedness. The Board plans to invite representatives from Nevada state and local governments, industry associations, and public interest groups to make presentations. A roundtable discussion will cover key topics raised during the presentations.

The afternoon session will be devoted to a discussion of performance assessment. The Board has asked for presentation on the DOE's newly drafted siting guidelines, 10 CFR 960, including the basis for the proposed revisions. The Board also would like to know about DOE plans to make the logic and reasoning that underlie performance assessment ''transparent'' to both scientific and lay communities.

Time has been set aside for public comment and questions on both days. To ensure that everyone wishing to speak is provided time to do so, the Board encourages those who have comments to sign the Public Comment Register, which will be located at the registration table. A time limit may have to be set on the length of individual remarks; however, written comments of any length may be submitted for the record.

The Nuclear Waste Technical Review Board was created by Congress in the Nuclear Waste Policy Amendments Act of 1987 to evaluate the technical and scientific validity of activities undertaken by the DOE in its program to manage the disposal of the Nation's spent nuclear fuel and defense high-level waste. In the same legislation, Congress directed the DOE to characterize a site at Yucca Mountain, Nevada, for its suitability as a potential location for a permanent repository for the disposal of that waste.

Transcripts of this meeting will be available via e-mail, on computer disk, or on a library-loan basis in paper format from Davonya Barnes, Board staff, beginning February 26, 1997. For further information, contact Frank Randall, External Affairs, 1100 Wilson Boulevard, Suite 910, Arlington, Virginia 22209; (Tel) 703–235–4473; (Fax) 703–235–4495.

Dated: November 22, 1996.

William Barnard,

Executive Director, Nuclear Waste Technical Review Board.

[FR Doc. 96–30882 Filed 12–3–96; 8:45 am]

BILLING CODE 6820–AM–M