the violations and proposed no reason for mitigating the civil penalties; rather, each officer maintained that he was not responsible for the violations and each officer proposed that the other officer should be held responsible for the violations and associated civil penalties.

Summary of the Licensee's Responses Concerning Liability and Responsibility for the Violations

1. PI's Response Dated May 13, 1997 (Submitted by Mr. Chambers, PI's Secretary/ Treasurer): Mr. Chambers protested the proposed civil penalties arguing that he is neither the owner nor President of PI, and that his involvement with PI was strictly as an investor. In addition, Mr. Chambers maintained that he did not take part in the day-to-day operations of PI and that Mr. Kumar, President and major stockholder of PI, is fully responsible for the violations. Mr. Chambers subsequently provided the NRC a copy of ''Stock Restriction and Purchase Agreement'' among PI, Mr. Chambers, and Mr. Kumar as evidence that his involvement was strictly as an investor.

2. PI's Responses Dated October 28, 1997, and January 6, 1998 (Submitted by Mr. Kumar, PI's President): Mr. Kumar's responses submitted by Mr. Manifesto, Mr. Kumar's counsel, argued that Mr. Chambers was the secretary/treasurer of PI during the relevant time period and that PI was owned jointly by Mr. Kumar and Mr. Chambers. Mr. Kumar further argued that Mr. Chambers had total control of the bank account of the corporation, and had equal financial control over all financial matters, as evidenced by the fact that no payment in excess of $1,000.00 could be made without Mr. Chambers' signature. In addition, Mr. Kumar maintained that: (1) Mr. Chambers served not only as an officer, but also on the Board of Directors of PI; and (2) after Mr. Kumar severed his relation with PI in August 1994, Mr. Chambers maintained all of the assets of PI, including bank accounts and equipment.

NRC Evaluation of the Licensee's Responses

The Licensee's arguments, as set forth above, do not provide a basis under the NRC's Enforcement Policy for mitigation or remission of the civil penalties. As to the question of responsibility, PI must pay the civil penalty in accordance with this Order. The Licensee's arguments do not relieve Mr. Chambers or Mr. Kumar of their responsibilities for ensuring that PI pays the civil penalty. Both Mr. Chambers and Mr. Kumar were part-owners and corporate officers of PI during the time period when the violations of NRC requirements occurred.

Therefore, after careful consideration of the responses, the NRC has determined that neither Mr. Chambers nor Mr. Kumar provided an adequate basis for the NRC to conclude that they should not be responsible for ensuring payment of the civil penalties by PI concerning its violations of NRC requirements. The NRC's determination is based on the fact that:

- Mr. Chambers served as an officer, and on the Board of Directors, of PI during the relevant time period; Mr. Chambers had control of all personnel matters during the relevant time period; Mr. Chambers had total financial control of PI; and Mr. Chambers maintained all of PI's assets, including bank accounts and equipment after PI became defunct.
- Mr. Kumar was the President of PI during the relevant time period; Mr. Kumar is the last known President of Power Inspection as noted in a July 16, 1996 ''Stock Restriction and Purchase Agreement''; Mr. Kumar is currently listed as the Chief Executive Officer of PI on the Pennsylvania Department of State Corporate/Limited Partnership records; and Mr. Kumar is currently listed as the Chief Executive Officer/President of PI on the Dunn & Bradstreet listing.

NRC Conclusion

The NRC has considered all of the arguments the Licensee made and concluded that the Licensee has not provided an adequate basis for mitigation of the proposed civil penalties. In addition, the NRC has concluded that Mr. Chambers and Mr. Kumar are responsible for ensuring payment of the civil penalties by PI concerning its violations of NRC requirements. Consequently, the civil penalties in the amount of $40,000 should be imposed by order.

[FR Doc. 98–3433 Filed 2–10–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Licensing Support System Advisory Review Panel; Meeting

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of Public Meeting.

SUMMARY: The Licensing Support System Advisory Review Panel (LSSARP) will hold its next meeting on February 24 and 25, 1998, in Las Vegas, Nevada. A future notice will specify the exact location for the meeting. The meeting will be open to the public pursuant to the Federal Advisory Committee Act (Pub. L. 94–463, 86 Stat. 770–776).

AGENDA: The meeting will be held from 8:30 a.m. to 4:30 p.m. on Tuesday, February 24, and from 8:30 a.m. to 10:00 a.m., as needed, on Wednesday, February 25, 1998. The purpose of the meeting is to discuss amendments proposed by the Nuclear Regulatory Commission (NRC) to its regulations concerning the design and operation of the Licensing Support System (LSS). The proposed amendments were published in the **Federal Register** on November 13, 1997 (62 FR 60789). The time period for comments on the proposed amendments expires on March 30, 1998.

SUPPLEMENTARY INFORMATION: The Nuclear Regulatory Commission (NRC) established the LSSARP in 1989 to provide advice and recommendations to the NRC and to the Department of Energy (DOE) concerning the design, development and operation of an electronic information management system, known as the Licensing Support System (LSS), for the storage and retrieval of information relevant to the Commission's future licensing proceeding for a geologic repository for the disposal of high-level radioactive waste. Membership on the panel consists of representatives of the State of Nevada, Nye County Nevada, a coalition of local counties of Nevada and California adjoining Nye County, the National Congress of American Indians, the Nevada Nuclear Waste Task Force, the nuclear industry, DOE, NRC and other agencies of the Federal government which have experience with large electronic information management systems.

FOR FURTHER INFORMATION CONTACT: John C. Hoyle, Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555: telephone 301–415–1969.

Public Participation: Interested persons may make oral presentations to the Panel or file written statements. Requests for oral presentations should be made to the contact person listed above as far in advance as practicable so that appropriate arrangements can be made.

Dated: February 5, 1998.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 98–3430 Filed 2–10–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the

Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 16, 1998, through January 30, 1998. The last biweekly notice was published on January 28, 1998 (63 FR 4308).

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 and Directives Branch, Division of Administration 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 March 13, 1998, 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: November 1, 1996.

Description of amendment request: The change increases the surveillance interval to allow verification that a reactivity anomaly does not exist to every 1100 MWD/T (megawatt-days per metric ton) average core exposure (approximately 41 days) instead of once every one effective full power month (approximately 30 days).

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?

This change increases the surveillance interval to allow verification that a reactivity anomaly does not exist every 1100 MWD/T average core exposure (approximately 41 days) instead of once every one effective full power month (approximately 30 days). Reactivity anomalies are not considered to be initiators of any analyzed event. Operating history has shown that the difference between predicted and monitored core reactivity is continually acceptable during the extended Surveillance interval. The consequences of an accident are not affected by relaxing the Frequency of the Surveillance since the consequences of an event with a reactivity anomaly during the current interval (due to not detecting the existence of a reactivity anomaly between Surveillances) are the same as the consequences of an event with a reactivity anomaly during the additional period. Additionally, the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. This change does not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change does not involve a significant increase in the probability or consequences of a previously analyzed accident.

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

The change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change is acceptable since the proposed Frequency is adequate for ensuring a reactivity anomaly does not exist. Operating history has shown that the difference between predicted and monitored core reactivity is continually acceptable during the extended Surveillance interval. Also, this change is considered acceptable since the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. The safety analysis assumptions will still be maintained, thus, no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The current Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP) only address a single inoperable scram accumulator, requiring entry into TS 3.0.3 for direction to shut down a unit if additional scram accumulators become inoperable. The proposed change corrects this situation by revising the declared status of control rods with inoperable scram accumulators and allowing a short out-of-service time for the control rod scram accumulators before requiring a unit shutdown, consistent with the Improved Technical Specifications (ITS) (NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 1, April 1995). In the event scram accumulators are inoperable concurrent with low charging water header pressure, the ITS require that the reactor mode switch be placed in the ''shutdown'' position, which ensures that all control rods are inserted and the unit is shutdown. The proposed change deviates from the ITS in that it requires a manual scram under these conditions which also ensures that all control rods are inserted and the unit is shutdown. Details associated with this deviation are included in a Carolina Power & Light Company letter dated September 11, 1997 (see response to NRC comment 3.1.5–2), which is available to the public.

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 revises the declared status of control rods with inoperable scram accumulators and allows a short out-of-service time for the control rod scram accumulators before requiring a plant shutdown. Inoperable scram accumulators are not considered initiators for any accidents previously evaluated, and therefore, cannot increase the probability of such accidents. The extended time period to declare a control rod inoperable provides a reasonable time to attempt investigation and restoration of the inoperable control rod scram accumulator. This time period is acceptable since the time period is sufficiently short such that it does not increase the risk significance of an ATWS [anticipated transient without scram] event. Furthermore, this change will add actions which will address the situation where multiple control rod scram accumulators may rapidly become inoperable. In addition, the change that allows modifying the status of a control rod with an inoperable scram accumulator is acceptable since the numbers and distribution of control rods are restricted and Technical Specification actions continue to ensure that the control rods can still perform their safety function when required. As a result, this change will not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not involve physical modification to the plant. The change in the operation is consistent with current safety analysis assumptions. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change is consistent with the assumptions of the current safety analysis. The extended time to evaluate and access two or more inoperable control rod scram accumulators and the allowance to declare any control rod with an inoperable scram accumulator ''slow'' when operating at a reactor pressure [greater than or equal to] 950 psig proposed by this change is acceptable since adequate controls are added to the Technical Specifications which ensure charging water header pressure to the control rod scram accumulators is maintained and action is provided to immediately shutdown the reactor before the scram safety function is significantly impacted in the event cha[r]ging water header pressure cannot be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed changes extend the refueling interval surveillance Frequencies that are currently specified as 18 months for surveillances other than those associated with instrumentation channel calibration to 24 months.

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 changes involve a change in the surveillance Frequency from 18 months to 24 months. The change in surveillance Frequency is not assumed to be an accident initiator for any accidents previously evaluated in the SAR [Updated Final Safety Analysis Report]. Therefore, this change will have no impact on the probability of an accident previously evaluated. By changing the Surveillance Frequency from 18 months plus grace to a maximum of 30 months, the consequences of an accident previously evaluated in the SAR are not significantly increased. This is based on the fact that the evaluation of the subject changes demonstrated that the overall impact, if any, on the systems['] availability is minimal. Since the impact on the systems is minimal, it can be concluded that the overall impact on the plant accident analysis is negligible. Furthermore, it is shown that the performance history for the subject systems does not indicate any failures which would invalidate the conclusions reached in this evaluation.

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

This proposed change will not involve any physical changes to plant systems, structures, or components. The changes in normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety has not been significantly reduced. Although, there will be an increase in the interval between the subject surveillance tests, the evaluation of the changes demonstrates that there is no evidence of any failures which would impact the subject systems['] availability. Based on the fact that the increased testing interval has a minimal impact on the subject systems, it can be concluded that the assumptions in the licensing basis are not impacted by the changes in the subject requirements and commitments.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change involves a change in the instrumentation channel calibration surveillance testing intervals from 18 months to 24 months.

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 involves a change in the instrumentation channel calibration surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other test[s] performed more frequently will identify potential equipment problems. Furthermore, a historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

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

The proposed change involves a change in the instrumentation channel calibration surveillance testing intervals from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, a historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is small based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change allows a short out-of-service time for various combinations of inoperable emergency core cooling system (ECCS) subsystems instead of an immediate plant shutdown.

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 allows a short out-of-service time for various combinations of inoperable ECCS subsystems instead of an immediate plant shutdown. ECCS equipment is used to mitigate the consequences of an accident, but the inoperability of ECCS equipment is not considered as the initiator of any previously analyzed accident. As such, the inoperability of ECCS subsystems will not increase the probability of any accident previously evaluated. The proposed combinations of inoperable ECCS subsystems are bounded by the analysis summarized in NEDC–31624P which utilizes an NRC [Nuclear Regulatory Commission] approved methodology for determining consequences. This analysis demonstrated that adequate core cooling would still be provided with the proposed change. Therefore, the consequences of an event occurring during the proposed allowed outage time are the same as the consequences of an event occurring during the current period allowed to place the plant in a shutdown condition. As a result, the change does not involve a significant increase in the consequences of any accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed combinations of inoperable ECCS subsystems are bounded by the analysis summarized in NEDC–31624P which utilizes an NRC approved methodology. This analysis demonstrated that adequate core cooling would still be provided with the proposed change. In addition, the allowable outage time specified is based on a reliability study (Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), ''Recommended Interim Revisions to LCOs [limiting conditions for operation] for ECCS Components,'' December 1, 1975) and has been found to be acceptable through operating experience. Any reduction in the margin of safety is offset by the benefit of reducing the transient risk associated with an immediate plant shutdown. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change reduces the number of automatic depressurization system (ADS) valves required to be OPERABLE from seven to six.

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 number of ADS valves required to be OPERABLE from seven to six. The number of ADS valves required to be OPERABLE is not assumed in the initiation of any analyzed event. Therefore, the change does not increase the probability of an accident previously evaluated.

The ADS valves function to mitigate the consequences of analyzed events by reducing the reactor vessel pressure to allow low pressure ECCS [emergency core cooling system] components to function as needed in the event of a HPCI [high-pressure coolant injection] System failure. The change is based on the analysis summarized in NEDC–31624P, ''Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis,'' Revision 2, July 1990. This analysis shows that adequate core cooling is provided during a small break LOCA and a simultaneous HPCI System failure (limiting LOCA) with two of the seven ADS valves out-of-service. NEDC–31624P was previously reviewed and accepted by the NRC [Nuclear Regulatory Commission] as documented in a letter from E.G. Tourigny (NRC) to L.W. Eury (CP&L), ''SAFER/GESTR–LOCA Analysis, Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. 72854/72855),'' dated 06/01/89 and a letter from E.G. Tourigny (NRC) to L.W. Eury (CP&L), ''Revision of SAFER/GESTR–LOCA Analysis—Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. 77585 and 77586),'' dated 01/10/91. The change is considered acceptable since the analyses show that only five ADS valves are required to perform the intended safety function of lowering reactor pressure. As a result, the change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not involve physical modification to the plant and the proposed change continues to provide assurance that the ADS can perform its intended safety function when required. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change does not involve a significant reduction in a margin of safety since sufficient ADS valves are maintained to ensure the safety analysis assumptions are met. The safety analysis shows that, with a HPCI failure, five ADS valves are sufficient to lower reactor pressure to allow low pressure ECCS injection and cooling. Thus, the proposed change does not impact the 10 CFR 50.46 limits. NEDC–31624P was previously reviewed and accepted by the NRC as documented in a letter from E.G. Tourigny (NRC) to L.W. Eury (CP&L), ''SAFER/GESTR–LOCA Analysis, Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. 72854/72855),'' dated 06/01/89 and a letter from E.G. Tourigny (NRC) to L.W. Eury (CP&L), ''Revision of SAFER/GESTR–LOCA Analysis—Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. 77585 and 77586),'' dated 01/10/91. As a result, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: This change will raise the minimum pressure at which the automatic depressurization system (ADS) is required to be OPERABLE to 150 psig.

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?

This change will raise the minimum pressure at which ADS is required to be OPERABLE to 150 psig. The OPERABILITY of the ADS valves below 150 psig is not assumed in the initiation of any analyzed event. The ADS is assumed in the mitigation of consequences of a LOCA [loss-of-coolant accident] which occurs at high reactor pressure. The ADS is not assumed in the mitigation of low reactor pressure events since its function is to lower the pressure to within the capabilities of the low pressure makeup systems. Low pressure injection systems are analyzed (per NEDC–31624P, ''Brunswick Steam Electric Plant Units

1 and 2 SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis,'' Revision 2, July 1990) to begin injection into the RPV [reactor pressure vessel] at pressures well above 150 psig. As a result, the proposed change does not impact the ability of the ECCS [emergency core cooling system] to perform [its] intended safety function and the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve physical modification to the plant and the proposed change continues to provide assurance that the ADS can perform its safety function when required. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The purpose of the ADS is to lower reactor pressure sufficiently to allow low pressure ECCS to inject and cool the core in the event of a HPCI [high-pressure coolant injection] System failure. Revising the minimum pressure for required ADS valve OPERABILITY is acceptable since the low pressure ECCS can provide core cooling at reactor pressures well above 150 psig and since the HPCI System is not required to be OPERABLE below 150 psig. As a result, 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.

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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: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change relaxes the low pressure emergency core cooling system (ECCS) pump flow acceptance criteria under operational conditions 1 (power operation), 2 (startup), and 3 (hot shutdown).

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 relaxes the low pressure ECCS pump flow acceptance criteria. Low pressure ECCS equipment is used to mitigate the consequences of an accident, but is not considered as the initiator of any previously analyzed accident. As such, the change does not increase the probability of any accident previously evaluated. The proposed low pressure ECCS pump flow acceptance criteria are assumed in the analysis summarized in NEDC–31624P [''Brunswick Steam Electric Plant Units 1 and 2 SAFR/GESTR–LOCA Loss-of-Coolant Accident Analysis,'' Revision 2, July 1990] which utilizes an NRC approved methodology for determining consequences. The resulting peak cladding temperature for all the cases analyzed in NEDC–31624P is below 1600 °F (a significant margin to the 10 CFR 50.46 limit). As a result, the ECCS subsystems assumed to be available during events analyzed will continue to provide adequate core cooling. Therefore, the change does not involve a significant increase in the consequences of any accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. In addition, the low pressure ECCS flow rates will not be determined in a new or different way. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed low pressure ECCS pump flow acceptance criteria are assumed in the analysis summarized in NEDC–31624P which utilizes an NRC approved methodology. NEDC–31624P concludes that the ECCS subsystems can still provide adequate core cooling with the proposed pump flow acceptance criteria and in all cases analyzed peak cladding temperature is maintained below 1600 °F. In addition, plant procedures will continue to trend the performance of the low pressure ECCS pumps and ensure that any adverse trends in equipment performance are identified and appropriate actions taken. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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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: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change relaxes the core spray (CS) pump flow acceptance criterion during shutdown conditions.

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

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

The proposed change relaxes the CS pump flow acceptance criterion. Low pressure ECCS [emergency core cooling system] equipment is used to mitigate the consequences of a reactor vessel draindown event during shutdown conditions, but is not considered as the initiator of any previously analyzed accident. As such, the change does not increase the probability of any accident previously evaluated. The proposed low pressure ECCS pump flow acceptance criteria are assumed in the analysis summarized in NEDC–31624P [''Brunswick Steam Electric Plant Units 1 and 2 SAFR/GESTR–LOCA Loss-of-Coolant Accident Analysis,'' Revision 2, July 1990] which utilizes an NRC approved methodology for determining consequences. The resulting peak cladding temperature for all the cases analyzed in NEDC–31624P is below 1600 °F (a significant margin to the 10 CFR 50.46 limit). This analysis assumes the reactor was operating at high power. This analysis did not invalidate the long term cooling analysis described in NEDO–20566A [''General Electric Company Analytical Model for Loss of Coolant Analysis in accordance with 10 CFR 50 Appendix K'']. Therefore, since the CS pump flow proposed by this change is adequate for high power conditions, it is reasonable to assume the CS pump flow is adequate to restore and maintain adequate vessel level during an inadvertent vessel draindown event while shutdown. The required low pressure ECCS subsystems during events analyzed in shutdown conditions will continue to provide adequate redundancy and coolant makeup capability. Therefore, the change does not involve a significant increase in the consequences of any accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. In addition, the CS pump flow rate will not be determined in a new or different way. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed CS pump flow acceptance criterion is assumed in the analysis summarized in NEDC–31624P which utilizes an NRC approved methodology. NEDC–31624P concludes that the ECCS subsystems can still provide adequate core cooling with the proposed CS pump flow acceptance criterion and in all cases analyzed peak cladding temperature is maintained below 1600 °F. Since the analysis assumed high power conditions, it is reasonable to assume that, with the proposed change, adequate coolant makeup capability is maintained during shutdown conditions. In addition, plant procedures will continue to trend the performance of the low pressure ECCS pumps and ensure that any adverse trends in equipment performance are identified and appropriate actions taken. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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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: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: This proposed change eliminates current Technical Specification (CTS) 3/4.6.1.5, Primary Containment Internal Pressure.

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

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

This proposed change eliminates CTS 3/4.6.1.5, Primary Containment Internal Pressure. This change does not result in any hardware or operating procedure changes. The primary containment pressure is not assumed to be an initiator of any analyzed event. It is an initial condition in the containment analysis (e.g., following a DBA LOCA [design-basis accident loss-of-coolant accident]). CTS 3/4.6.1.5 was necessary to maintain this assumption which helps ensure that the primary containment design pressure is not exceeded following an accident. However, the power uprate analysis modified this initial drywell pressure value such that the assumed value is greater than the RPS [reactor protection system] high drywell trip. The results of the power uprate analysis show that this modified initial drywell pressure is acceptable for ensuring primary containment pressure design limits are not exceeded. This modified initial pressure was utilized in determining a new P_a [calculated peak containment internal pressure related to the design basis accident], and has been submitted to the NRC to support the BNP [Brunswick Nuclear Plant] power uprate amendment.

The initial drywell pressure assumption is being ensured by the RPS high drywell pressure scram, which will trip the unit prior to exceeding the assumed drywell pressure value, effectively placing the unit in MODE 3. While the RPS trip is not required in MODE 3, the Emergency Operating Procedures (EOPs) will govern actions if the drywell pressure exceeds the assumed drywell pressure value. The EOPs will require entry into the Reactor Vessel Control and Primary Containment Control actions. These actions require steps to reduce primary containment pressure to below the value assumed in the accident analyses and to cool down the reactor at normal cooldown rates to MODE 4 if pressure cannot be reduced below the reactor trip setpoint. The negative pressure limit is controlled and met by the design and proper operation of the reactor building-to-suppression chamber and the suppression chamber-to-drywell vacuum breakers. These vacuum breakers, which are required to be OPERABLE in MODES 1, 2, and 3, are designed to ensure the negative pressure design limit of the primary containment is not exceeded. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not require physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No significant reduction in a margin of safety is involved. The upper pressure limit is maintained by the design and proper operation of the RPS high drywell pressure trip, a Technical Specification required instrumentation function, and the EOPs. The negative pressure limit is being maintained by the design and proper operation of the reactor building-to-suppression chamber and suppression chamber-to-drywell vacuum breakers, also Technical Specification required components. Therefore, adequate controls exist with respect to the primary containment pressure limits to ensure the primary containment pressure will not be exceeded in the event of a design basis event.

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

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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: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change relocates requirements and surveillances for the Containment Air Dilution (CAD) system from the Technical Specifications to a licensee controlled document. Licensee analysis has demonstrated that the CAD system is not needed to maintain the primary containment atmosphere below flammability limits.

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

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

The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the BNP [Brunswick Nuclear Plant] Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be permitted.

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.

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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: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change applies to the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, and provides longer out-of-service times for various combinations of inoperable service water (SW) pumps and deletes various limitations of which pumps can be inoperable.

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 provides longer out-of-service times for various combinations of inoperable SW pumps and deletes various limitations of which pumps can be inoperable (*e.g.*, a remaining unit specific NSW [nuclear service water] pump must be electrically separated from the remaining CSW [conventional service water] pump). The SW System supports safety related systems used to mitigate the consequences of an accident, but the inoperability of the SW System is not considered as the initiator of any previously analyzed accident. As such, the inoperability of SW pumps will not increase the probability of any accident previously evaluated. The proposed combinations of inoperable SW pumps are bounded by the analyses summarized in CP&L calculations PCN GOO50A–10 [''BSEP Unit No. 1 Service Water System Hydraulic Analysis,'' Revision 6, dated July 29, 1993] and PCN GOO50A–12 [''BSEP Unit No. 2 Service Water System Hydraulic Analysis,'' Revision 5, dated August 11, 1992] which have been previously evaluated by the NRC. These analyses demonstrate that adequate SW cooling capability would still be provided with the proposed changes. Therefore, the consequences of an event occurring during the proposed allowed outage times are the same as the consequences of an event occurring during the current allowed outage time period or the current period allowed to place the plant in a shutdown condition. As a result, the change does not involve a significant increase in the consequences of any accident previously evaluated.

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

The proposed change does not involve physical modification to the plant or changes in parameters governing normal plant operation. The proposed change continues to provide assurance that the SW System is capable of performing its required support function. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed combinations of inoperable SW pumps are bounded by the analyses summarized in CP&L calculations PCN GOO50A–10 and PCN GOO50A–12 which have been previously evaluated by the NRC. These analyses demonstrate that adequate SW cooling capability would still be provided with the proposed change. In addition, the proposed allowable outage times and the capability of the SW System to support additional single failures are consistent with the allowable outage times and capability of other safety related systems with similar levels of degradation. Any reduction in the margin of safety is offset by the benefit of reducing the transient risk associated with an unnecessary plant shutdown. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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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: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change allows the extension of the Allowed Outage Time (AOT) from 24 hours to 7 days of a shutdown unit's 4.16 kilovolt (kV) balance of plant (BOP) bus which is needed to support loads required by the operating unit.

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?

Extending AOT of a shutdown unit's BOP bus from 24 hours to 7 days will not increase the probability of occurrence of an accident on the operating unit. The probability of a previously evaluated accident would not be increased by the longer AOT since de-energization of a single BOP bus is not considered in the initiation of any previously analyzed event. The BOP buses support the distribution of offsite power to the Class 1E AC Electrical Power Distribution System, which supports equipment necessary for the mitigation of accidents. Extending the AOT of a shutdown unit's BOP bus will not significantly increase the consequences of an accident on the operating unit. The consequences of an accident occurring during the proposed 7 day AOT would be the same as the consequences associated with the existing 24 hour AOT. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is defined by the scenario where a LOCA [loss-of-coolant accident] occurs on the operating unit concurrent with loss of offsite power and the worst case single failure (e.g., loss of a DG [diesel generator] and associated supported loads). The intentional de-energization of one of the AC Electrical Power Distribution System load groups primarily associated with the shutdown unit, as a result of de-energization of a BOP bus associated with the shutdown unit, will leave three AC Electrical Power Distribution System load groups OPERABLE each with their associated emergency diesel generator and two sources of offsite power OPERABLE. Two of these AC Electrical Power Distribution System load groups will be associated with the operating unit and one with the shutdown unit. Loss of an AC Electrical Power Distribution System load group primarily associated with the shutdown unit is not as limiting to the operating unit as the loss of one of its emergency power system load groups; there are fewer operating unit loads required for mitigation of accident and transients affected by the removal of an AC Electrical Power Distribution System load group primarily associated with the shutdown unit. The intentional de-energization of an AC Electrical Power Distribution System load group primarily associated with the shutdown unit, as a result of de-energization of a BOP bus, is enveloped by the LOCA scenario described above.

There are a number of operating unit loads required for mitigation of accidents and transients which will become inoperable when an AC Electrical Power Distribution System load group primarily associated with the shutdown unit is removed from service as a result of de-energization of the associated BOP bus. A review of the loads supported by each of the load groups indicates that operating unit loads required for mitigation of accidents and transients can either be supplied from an alternate source or the Technical Specifications would allow an AOT of 7 days or greater for the affected loads. Changing the AOT from 24 hours to 7 days for an inoperable BOP bus associated with the shutdown unit would not exceed the AOT for these individual loads. In addition, operating unit primary containment isolation valves supplied from the shutdown unit's out of service load group (RHR [residual heat removal] Outboard Injection, RHR Inboard Injection, and RHR Torus Spray) would be closed, in accordance with the Technical Specification requirements of the operating unit, to ensure they perform their safety function if needed. The proposed AOT for an inoperable BOP bus associated with [the] shutdown unit provides the benefit of improved reliability and availability of the AC Electrical Power Distribution System and the associated offsite power circuits (via upstream BOP buses) since the longer AOT will allow maintenance of the buses of these load groups to be performed on a more optimum schedule. As a result, the proposed change does not involve a significant decrease 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.

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NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change allows extension of the Allowed Outage Time (AOT) from 8 hours to 7 days of one of the shutdown unit's emergency load groups which is needed to support loads required by the operating unit.

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?

Extending the Allowed Outage Time (AOT) of an AC Electrical Power Distribution System load group primarily associated with a shutdown unit from 8 hours to 7 days will not increase the probability of occurrence of an accident on the operating unit. The probability of a previously evaluated accident would not be increased by the longer AOT since de-energization of a single load group is not considered in the initiation of any previously analyzed event. The Class 1E AC Electrical Power Distribution System supports equipment necessary for the mitigation of accidents. Extending the AOT of an AC Electrical Power Distribution System load group primarily associated with a shutdown unit will not significantly increase the consequences of an accident on the operating unit. The consequences of an accident occurring during the proposed 7 day AOT would be the same as the consequences associated with the existing 8 hour AOT. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is defined by the scenario where a LOCA [loss-of-coolant] occurs on the operating unit concurrent with loss of offsite power and the worst case single failure (e.g., loss of a DG [diesel generator] and associated supported loads). The intentional de-energization of one of the AC Electrical Power Distribution System load groups primarily associated with the shutdown unit will leave three AC Electrical Power Distribution System load groups OPERABLE each with their associated emergency diesel generator and two sources of offsite power OPERABLE. Two of these AC Electrical Power Distribution System load groups will be associated with the operating unit and one with the shutdown unit. Loss of an AC Electrical Power Distribution System load group primarily associated with the shutdown unit is not as limiting to the operating unit as the loss of one of its emergency power system load groups; there are fewer operating unit loads required for mitigation of accident and transients affected by the removal of an AC Electrical Power Distribution System load group primarily associated with the shutdown unit. The intentional de-energization of an AC Electrical Power Distribution System load group primarily associated with the shutdown unit is enveloped by the LOCA scenario described above.

There are a number of operating unit loads required for mitigation of accidents and transients which will become inoperable when an AC Electrical Power Distribution System load group primarily associated with the shutdown unit is removed from service. A review of the loads supported by each of the load groups indicates that operating unit loads required for mitigation of accidents and transients can either be supplied from an alternate source or the Technical Specifications would allow an AOT of 7 days or greater for the affected loads. Changing the AOT from 8 hours to 7 days for an inoperable AC Electrical Power Distribution System load group primarily associated with a shutdown unit would not exceed the AOT for these individual loads. In addition, operating unit primary containment isolation valves supplied from the shutdown unit's out of service load group (RHR [residual heat removal] Outboard Injection, RHR Inboard Injection, and RHR Torus Spray) would be closed, in accordance with the Technical Specification requirements of the operating unit, to ensure they perform their safety function if needed. The proposed AOT for an inoperable AC Electrical Power Distribution System load group provides the benefit of improved reliability and availability of the AC Electrical Power Distribution System since the longer AOT will allow maintenance of the buses of these load groups to be performed on a more optimum schedule. As a result, the proposed change does not involve a significant decrease 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.

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NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change allows reactor coolant system (RCS) hydrostatic pressure and leakage testing to be performed with average reactor coolant temperature in excess of 212°F and not consider the plant to be in MODE 3 (hot shutdown) provided certain conditions are met.

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 allows RCS hydrostatic pressure and leakage testing to be performed with average reactor coolant temperature in excess of 212°F and not consider the plant to be in MODE 3 provided certain conditions are met. The probability of a leak or a pipe break in the reactor coolant pressure boundary during inservice leak and hydrostatic testing is not increased by allowing reactor coolant temperature to exceed 212°F because the Reactor Coolant System is designed for temperatures exceeding 500°F with similar pressures. In addition, because an inspection is being performed on the Reactor Coolant System piping while it is being pressurized, the probability of a crack going unnoticed and resulting in a pipe break is reduced. Reactor vessel integrity will not be compromised by performing hydrostatic pressure and leakage testing at temperatures in excess of 212°F. Performing hydrostatic pressure and leakage testing above 212°F would allow steam, rather than water to emit from a leak or pipe break. The hydrostatic or inservice leak test is performed with a water solid reactor pressure vessel. An engineering analysis was performed to determine the reactor building pressure and temperature effects if a pipe break occurred during the hydrostatic pressure and inservice leak testing at a reactor coolant temperature of 275°F. A recirculation line break was used in the analysis since it was considered the most conservative pipe break with primary containment breached during the test. This analysis has concluded that the recirculation line break during the performance of the test could result in a rise in reactor building pressure sufficient to cause the opening of the reactor building blowout panel and result in a breach of secondary containment. Furthermore, this analysis has shown without credit for HVAC [heating, ventilation, and air conditioning] operation, there would also be a short term increase in the reactor building ambient temperature. However, when compared to the UFSAR [Updated Final Safety Analysis Report] LOCA [loss-of-coolant accident] analysis and the UFSAR main steam line break analysis, it can be concluded that the consequences relative to offsite doses, reactor building pressures and temperatures are bounded by previously analyzed accidents. This change will require that secondary containment be OPERABLE and capable of handling airborne radioactivity from steam leaks that could occur during the performance of hydrostatic pressure or inservice leak testing. Requiring secondary containment to be OPERABLE will conservatively ensure that, in the absence of a pipe break, potential airborne radiation from steam leaks will be filtered through the Standby Gas Treatment System, thereby minimizing radiation releases to the environment. Leaks to secondary containment would typically be detected by leakage inspections before significant inventory loss occurred. This is an integral part of the hydrostatic pressure and inservice leak testing program. In addition, there is no mechanism to impart additional fission products into the reactor coolant. Since the hydrostatic pressure test is performed after refueling, few noncondensible gases remain in the reactor coolant. In the proposed condition, the stored energy in the reactor core will be the same as that at 212°F. This stored energy is sufficiently low such that even with the loss of inventory following a recirculation line break, the core coverage could be maintained and the fuel would not exceed its peak clad temperature limit. Therefore, no significant release of fission products would occur. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical changes to plant structures, systems, or components (no new or different type of equipment will be installed and no equipment will be removed). The change will not alter assumptions made in the safety analyses. Therefore, the change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change allows RCS hydrostatic pressure and leakage testing to be performed with average reactor coolant temperature in excess of 212° F and not consider the plant to be in MODE 3 provided certain conditions are met. Secondary containment will be required to be maintained during the test and all required systems with the reactor in MODE 4 [cold shutdown] will be OPERABLE in accordance with the Technical Specifications. Since the hydrostatic or leak tests are performed water solid, at low decay heat values, and near MODE 4 conditions, the stored energy in the reactor core will be very low. Under these conditions, the potential for failed fuel and a subsequent increase in coolant activity is minimized. The reactor pressure vessel would rapidly depressurize in the event of a large primary system leak and the low pressure injection systems normally OPERABLE in MODE 4 would be adequate to keep the core flooded. This would ensure that the fuel would not be uncovered and would not exceed the 2200° F peak clad temperature limit. Moreover, requiring secondary containment, including isolation capability, to be OPERABLE will assure that potential airborne radiation from small leaks can be filtered through the Standby Gas Treatment System. This will ensure that doses remain within the limits of 10 CFR 100 guidelines. The potential doses from any leak or pipe break during the test are bounded by design basis accident doses presented in the UFSAR. Small system leaks would be detected by inspections before significant inventory loss has occurred. In addition, the change provides the benefit of avoiding depressurization and repressurization of the reactor pressure vessel during system hydrostatic or leakage pressure tests because of the lack of sufficient margin to the MODE 4/MODE 3 reactor coolant temperature transition limit. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change adds explicit exceptions to 10 CFR 50 Appendix J in the primary containment leakage testing program which were previously approved by the Nuclear Regulatory Commission for the Brunswick Steam Electric Plant Units 1 and 2.

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

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

The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: The proposed change would change the requirement of the Rod Block Monitor (RBM) to be Operable when Thermal Power is greater than or equal to 29% of Rated Thermal Power and less than 90% of the Rated Thermal Power with the minimum critical power ratio (MCPR) less than 1.70.

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 provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: November 1, 1996.

Description of amendment request: A Rod Worth Minimizer (RWM) CHANNEL FUNCTIONAL TEST is currently required to be performed during both a shutdown and a startup. The amendment request would modify the test frequency to require that the CHANNEL FUNCTIONAL TEST only be performed once provided the last test performance occurred within a 92-day period.

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?

CTS [Current Technical Specification] 4.1.4.1.1 requires a CHANNEL FUNCTIONAL TEST to be performed prior to withdrawal of control rods for the purpose of making the reactor critical and when the RWM is initiated during a plant shutdown. ITS [Improved TS] Surveillance Requirements are similar to CTS 4.1.4.1.1 except a test Frequency is specified (92 days). The proposed change effectively extends a[n] RWM Surveillance Frequency, i.e., the CHANNEL FUNCTIONAL TEST is not required to be performed if a startup or shutdown occurs within 92 days of a previous startup or shutdown. The RWM and associated Surveillance Requirements are not assumed as initiators of any previously analyzed accidents. In addition, operating history has shown that the RWM would be continually reliable during the extended Surveillance interval. The consequences of an accident are not affected by relaxing the Frequency of the Surveillance since the consequences of a design basis accident with the RWM inoperable during a reactor startup or shutdown (due to an undetected failure) are the same as the consequences of a design basis accident with the RWM inoperable for the proposed 92 day period. Additionally, the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. This change does not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

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

The change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to the Frequency is acceptable since the ITS Surveillance Frequency is adequate for ensuring the RWM is maintained OPERABLE.

Operating history has shown that the RWM would be continually reliable during the extended Surveillance interval. The most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. Also, the proposed change provides a benefit of eliminating unnecessary testing prior to startup and during a shutdown which reduces wear on the instruments, thereby increasing overall reliability. As such, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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: December 16, 1997.

Description of amendment request: The amendment request proposes to revise the Technical Specifications for the Shearon Harris Nuclear Plant. Specifically, the amendment request proposes revisions to TS 4.7.1.2.1.a.2.a, Auxiliary Feedwater System Surveillance Requirements, to change the differential pressure and flow requirements of the steam turbine-driven Auxiliary Feedwater (AFW) pump to allow testing of the pump at a lower speed than is currently performed.

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.

Changing the recirculation flow test parameters at which the turbine-driven AFW pump is tested will demonstrate pump operability while allowing the surveillance to be performed at a speed that is less detrimental to the pump. Appropriate testing will continue to ensure that the Auxiliary Feedwater System (AFS) is capable of performing its intended function. The proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Changing the recirculation flow test parameters at which the turbine-driven AFW pump is tested will demonstrate pump operability while allowing the surveillance to be performed at a speed that is less detrimental to the pump. Appropriate testing will continue to ensure that the AFS is capable of performing its intended function. The proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS.

The proposed amendment will not create any new accident scenarios, because the change does not introduce any new single failures, adverse equipment or material interactions, or release paths. 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.

Changing the recirculation flow test parameters at which the turbine-driven AFW pump is tested will demonstrate pump operability while allowing the surveillance to be performed at a speed that is less detrimental to the pump. Appropriate testing will continue to ensure that the AFS is capable of performing its intended function. 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: William M. Dean.

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

Date of amendment request: December 12, 1997.

Description of amendment request: The proposed amendments would modify the bypass logic for Main Steam Line Isolation Valve Isolation Actuation Instrumentation on Condenser Low Vacuum as stated in Technical Specification (TS) Tables 3.3.2–1 and 4.3.2.1–1.

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

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

The reactor vessel steam dome pressure switches, which are proposed to be removed from the Main Steam Isolation Valve (MSIV) closure scram bypass logic and the Condenser Vacuum—Low MSLIV [main steam line isolation valve] isolation bypass logic cause the above trip functions to become active when the reactor mode switch is not in the RUN position and the reactor pressure is greater than 1043 psig. The setpoints of the reactor vessel steam dome pressure switches are the same as the reactor vessel steam dome pressure—high scram function. Also, any pressure transients as a result of MSIV closure when not in Operational Condition 1, Run mode, are minor due to low steam flow compared to the same event at rated power. Therefore, the reactor pressure switches being removed from the bypass logic of the MSIV closure scram has little or no affect on reactor startup, operation, shutdown, or analyzed accidents.

The condenser vacuum—low isolation function bypass is interlocked by the same pressure switches that bypass the MSIV closure scram when the reactor mode switch is not in the RUN position. In addition to reactor pressure not high, the bypass of the condenser vacuum—low is bypassed only if the reactor mode switch is not in the RUN position, all Turbine Stop Valves (TSVs) are not full open, and the keylock bypass switches are in BYPASS (one for each channel).

With the reactor pressure interlock removed, the remaining interlocks assure that the condenser will not be overpressurized in Operational Conditions 2 and 3. The Reactor mode switch interlock limits reactor thermal power to less than about 12 percent in Operational Condition 2 (Control Rod withdrawal block on APRM [average power range monitor] High setpoint in Operational Conditions 2 and 5) and to much less than 1 percent power when all control rods are fully inserted in Operational Condition 3 after initial thermal power decay due to decay heat following reactor shutdown. The Turbine bypass valves can not be opened with condenser vacuum low (approximately the same as the isolation setpoint, but different instrumentation). The TSVs remain closed with condenser vacuum low due to a turbine trip on low condenser vacuum. Therefore, the remaining bypass interlocks assure that the isolation of the main steam lines will occur when needed to prevent overpressurization of the main condenser when vacuum is low or gone.

The change to the position information in the TS Table notes for the TSV bypass interlock corrects misinformation in the TS. The design has always used contacts from the auxiliary relays associated with the ''not-full-open'' limit switches for the MSIV closure scram. Therefore, the setpoints are the same as the MSIV closure scram in TS 2.2.1. The setpoint in the notes * are made approximate to avoid conflict with the RPS [reactor protection system] setpoints, which are controlling. Also, [sic] surveillances for the RPS function for TSV closure scram will continue to be performed per TS 4.3.1 at the frequencies specified in TS Table 4.3.1.1–1.

The setpoint for the TSV interlock is not a critical parameter for the isolation bypass interlock, since the normal position of the TSVs with low condenser vacuum is fully closed. Therefore, the use of an approximate value is sufficient, since the actual setpoints and surveillances are controlled by other specifications.

The reactor pressure switches being removed from the above bypass circuits are not used for the mitigation of any analyzed accidents or transients and may actually [decrease] the probability of a scram or isolation in Startup mode due to the potential for misoperation. Also, the correction to the TSV position in the bypass notes is more consistent with the actual setpoints, which are controlled by the Limiting Safety System Settings for RPS trip function due to TSV closure.

The rewording of Note * in TS Table 4.3.2.1–1 to be more like Note * in TS Table 3.3.2–1 helps avoid confusion due to wording differences and is an administrative type change.

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

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

The removal of the reactor pressure switches from the bypass logic for the MSIV closure scram function and the condenser vacuum—low MSLIV isolation function with a setpoint equal to the reactor pressure scram setpoint is not a significant change and does not alter the reactor modes in which the trips are or can be bypassed. When not in RUN mode, energy levels are low compared to events that could occur at rated power levels. These pressure switches only slightly change the bypass logic and do not affect the scram and isolation circuitry such that a new or different kind of accident would occur.

The correction of the TSV position interlock for the bypass function for the condenser vacuum—low MSLIV isolation is not a physical change to the plant, so no failure modes are affected or created.

The rewording of Note * in TS Table 4.3.2.1–1 to be more like Note * in TS Table 3.3.2–1 helps avoid confusion due to wording differences and is an administrative type change.

Therefore, the possibility of a new or different kind of accident is not created.

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

The removal of the rector pressure switches from the bypass logic of the MSIV closure scram function and the bypass logic from the condenser vacuum—low MSLIV isolation function does not reduce the margin of safety, because the setpoints were not established from analyses that have been performed. The setpoints were set at the value of the reactor scram on high reactor pressure as a convenient setpoint out of the way of normal plant operation, rather than initially removing the bypass interlock.

Also, the high reactor pressure scram is required to be operable in Operational Conditions 1, 2, and 3, and has no installed means of bypass, so the removal of the MSIV closure scram in Operational Conditions other than mode 1, Run mode becoming active due to high reactor pressure does not reduce the margin for reactor pressurization events.

The remaining bypass interlocks, associated with TSV position for the bypass of the condenser vacuum—low MSLIV isolation, assure that the main condenser will be protected from overpressurization events with low condenser vacuum. The TSVs are closed due to a main turbine trip with low condenser vacuum, so if the TSVs were to fail open, the MSLIV will occur in Operational Conditions 2 and 3 when required. The removal the reactor pressure bypass interlock and the correction to the TSV position will not be a significant reduction in the margin of safety.

The rewording of Note * in TS Table 4.3.2.1–1 to be more like Note * in TS Table 3.3.2–1 helps avoid confusion due to wording differences and is an administrative type change.

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

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: December 11, 1997.

Description of amendment request: The licensee proposed to revise Table 3.3–4 of the units' Technical Specifications, changing the Nuclear Service Water System Suction Transfer (from Lake Wylie to the Standby Nuclear Service Water Pond (SNSWP)) to a higher level of Lake Wylie. The Nuclear Service Water System is the ultimate heat sink for various heat loads during normal operation and design basis accidents. The system also provides makeup water to various systems. Lake Wylie provides the normal water supply whereas the SNSWP provides an assured water source should Lake Wylie water becomes unavailable. The transfer of suction is currently required to occur automatically when Lake Wylie's levels drops to an elevation of 552.9 feet. The proposed revision would change this requirement to a more conservative level about 2.5 feet higher than the current level. This change would correct previously identified nonconservative aspects of the net positive suction head (NPSH) calculation for the Nuclear Service Water System pumps.

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 analysis is presented below.

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

No. The revised suction transfer point would increase reliability of the Nuclear Service Water System by increasing the NPSH available to the system. No previously analyzed accidents were initiated by transfer of the suction source, and the transfer of suction was not a factor in the consequences of previously analyzed accidents. Therefore, the proposed change will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

No. Other than requiring suction be transferred at a higher level of Lake Wylie, the proposed change would not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

No. Margin of safety is associated with confidence in the design and operation of the plant. The proposed change to the Technical Specifications does not involve any change to plant design or operation. Thus, the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: December 18, 1997; revised on January 26, 1998.

Description of amendment request: The licensee proposed to revise the units' facility operating licenses (FOL) NPF–35 and NPF–52 to delete license conditions which have been fulfilled, to update information to reflect current plant status and regulatory requirements, and to make other editorial corrections. All the requested changes are administrative.

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 the FOL involves administrative changes only. No actual plant equipment, operating practices, or accident analyses are affected by this proposed amendment. Therefore, the proposed amendment has no impact on the possibility (sic) of any type of accident: new, different, or previously evaluated.

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

No. The proposed amendment to the Catawba FOL involves administrative changes only. No actual plant equipment, operating practices, or accident analyses are affected by this proposed amendment and no failure modes not bounded by previously evaluated accidents are created. Therefore, the proposed amendment has no impact on the possibility (sic) 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 license amendment is administrative in nature and only updates the Catawba FOL to eliminate outdated or completed requirements; therefore, no reduction in any existing margin of safety is involved.

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

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

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: December 12, 1997, with supplement dated August 13, 1997.

Description of amendment request: The proposed amendment establishes an alternate repair criteria for the segment of steam generator tubes that are located within the upper tube sheet.

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 Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The steam generators are used to remove heat from the reactor coolant system during normal operation and during accident conditions. The steam generator tubing forms a substantial portion of the reactor coolant pressure boundary. A steam generator tube failure is a violation of the reactor coolant pressure boundary and is a specific accident analyzed in the ANO–1 Safety Analysis Report.

The purpose of the periodic surveillance performed on the steam generators in accordance with ANO–1 Technical Specification 4.18 is to ensure that the structural integrity of this portion of the reactor coolant system (RCS) will be maintained. The technical specification plugging limit of 40% of the nominal tube wall thickness requires tubes to be repaired or removed from service because the tube may become unserviceable prior to the next inspection. Unserviceable is defined in the TS as the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an operating basis earthquake, a loss-of-coolant accident, or a steam line break.

The proposed technical specification specifies an alternate plugging limit for upper tubesheet volumetric outer diameter intergranular attack (ODIGA) indications. Based upon extensive testing and plant experience, it has been determined that upper tubesheet volumetric ODIGA flaws with a bobbin voltage indication less than that specified by the proposed technical specification can remain in service while maintaining the serviceability of the tube.

From testing performed on simulated flaws within the tubesheet, it has been shown that the patch IGA indications within the upper tubesheet, with depths up to 100% through-wall, do not represent structurally significant flaws which would increase the probability of a tube failure beyond that currently assumed in the ANO–1 Safety Analysis Report. The dose consequences of a MSLB accident are analyzed in the ANO–1 accident analysis. This analysis assumes the unit is operating with a 1 gpm steam generator tube leak and that the unit has been operating with 1% defective fuel. Increased leakage during a postulated MSLB accident resulting from applying the voltage-base repair criteria to upper tubesheet volumetric ODIGA is not expected. ODIGA has been present in the ANO–1 steam generators for many years with no known leakage attributed to this damage mechanism. Because of its localized nature and morphology, the flaw does not open under accident conditions. To further support this conclusion, hot leak testing at the bounding MSLB temperature, pressure, and load was performed on tubing with representative laboratory generated flaws. The leak testing was performed on 29 samples with volumetric ODIGA with bobbin indications of 0.04 to 1.62 volts. None of these flaws showed signs of leakage as a result of these loads. Additionally, four specimens created by electrodischarge machining (EDM) with depths up to approximately 95% through-wall were tested with no leakage detected. It was, therefore, concluded that volumetric ODIGA flaws with an eddy current indication up to 1.62 volts will not leak under accident conditions, and that this is an acceptable threshold value to use to assume zero accident leakage.

This change allows volumetric ODIGA flaws within the tubesheet, which are not projected to meet or exceed the 1.62 volt threshold when considering eddy current uncertainty and an allowance for growth, to remain in service. Continued operation with these flaws present does not result in a significant increase in the probability or consequences of an accident previously evaluated for ANO–1.

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

2. Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The steam generators are passive components. The intent of the technical specification surveillance requirements are being met by this change in that adequate structural and leakage integrity will be maintained. Additionally, the proposed change does not introduce any new modes of plant operation.

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

3. Does Not Involve a Significant Reduction in the Margin of Safety.

The margin of safety is not reduced by the implementation of the proposed technical specification change allowing volumetric ODIGA flaws within the upper tubesheet which meet the proposed acceptance criteria to remain in service.

Testing of upper tubesheet volumetric ODIGA flaws removed from the ANO–1 OTSGs during 1R13, showed the flawed tubes to be capable of withstanding differential pressures of 10,000 psid without the presence of the tubesheet. Testing of simulated through-wall flaws of up to 0.5 inch in diameter within a tubesheet showed that the tubes always failed outside of the tubesheet. Thus the structural requirements listed in the bases of the technical specification are satisfied considering this change.

Tubes with volumetric ODIGA indications within the tubesheet which satisfy the acceptance criteria specified in the proposed technical specification change are not anticipated to leak under accident conditions. This is due to the small size of the flaws and their morphology. This premise has been demonstrated through years of actual plant operation with no known leakage attributable to these flaws, even considering a plant transient in 1996 which exposed the ''B'' steam generator to a primary-to-secondary pressure differential of 2100 psid. The potential for leakage under accident conditions was the focus of testing performed on representative samples of flawed OTSG tubing. These tests confirmed for tubesheet flaws, within the bounds of the proposed technical specification change, that leakage is not expected under accident conditions. With no increased accident leakage anticipated as a result of the proposed technical specification change, the offsite dose consequences from a MSLB accident remain unchanged from that currently analyzed in the ANO–1 Safety Analysis Report.

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

In conclusion, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John Hannon.

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

Date of amendment request: November 18, 1996, as supplemented by letter dated January 21, 1998.

Description of amendment request: The amendment requests a change to Technical Specification (TS) Surveillance Requirement 4.4.8.3.1.b to test the Shutdown Cooling System suction line relief valves in accordance with TS 4.0.5. Editorial changes to 4.4.8.3.1 and 4.4.8.3.1.a. have also been requested.

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

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

No. The proposed change will not affect the assumptions, design parameters, or results of any accident previously evaluated. The proposed change does not add or modify any existing equipment. The proposed change will not diminish the ability of the valves to perform as required during an accident. The proposed Shutdown Cooling System suction line relief valves testing schedule will be in accordance with Section XI of the ASME.

Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR [Part] 50, Section 50.55a(g). This ensures the operational readiness of the valves. Therefore, the proposed change will not involve an increase in the probability or consequences of any accident previously evaluated.

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

No. The proposed change does not involve modifications to any existing equipment. The proposed change will not affect the operation of the plant or the manner in which the plant is operated. No new failure modes that have not been previously considered will be introduced. The net effect of the change is to allow the plant staff the option of reducing the frequency of valve testing to a level that has been acknowledged as acceptable by the applicable ASME Code. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change does not involve a decrease in the number or capacity of the valves in the system, nor does it involve a change in the relief valve setpoints, operability requirements, or limiting conditions for operation. The margin of safety for the relief valves is, in part, preserved by compliance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR [Part] 50, Section 50.55a(g). Although the proposed change will allow a slightly longer testing frequency, the proposed change will continue to preserve compliance with 10 CFR [Part] 50, Section 50.55a(g). Therefore, the proposed change will not involve a reduction in a margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: December 29, 1997.

Description of amendment request: The licensee proposed to modify specifications for selected cycle-specific reactor physics parameters so that they refer to the St. Lucie Unit 2 Core Operating Limits Report (COLR) for limiting values. Minor administrative changes are also included. The proposed Technical Specification (TS) changes utilized the guidance provided in Generic Letter 88–16 and are intended to be consistent with the Standard Technical Specifications for Combustion Engineering Plants (NUREG–1432, 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) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment relocates the calculated values of selected cycle-specific reactor physics parameter limits from the TS to the COLR, and includes minor editorial changes which do not alter the intent of stated requirements. The amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Parameter limits specified in the COLR for this amendment are not changed from the values presently required by Technical Specifications. Future changes to the calculated values of such limits may only be made using NRC approved methodologies, must be consistent with all applicable safety analysis limits, and are controlled by the 10 CFR 50.59 process. Assumptions used for accident initiators and/or safety analysis acceptance criteria are not changed by this amendment. Therefore, operation of the facility in accordance with the proposed amendment will 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 amendment relocates the calculated values of cycle specific reactor physics limiting parameters to the COLR and will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design configuration of St. Lucie plant systems. Therefore, 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.

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

The cycle specific parameter limits being relocated to the COLR by this amendment have not been changed from the values presently required by the TS, and a requirement to operate the plant within the bounds of the limits specified in the COLR is retained in the individual specifications. Future changes to the calculated values of these limits by the licensee may only be developed using NRC-approved methodologies, must remain consistent with all applicable plant safety analysis limits addressed in the Final Safety Analysis Report (FSAR), and are further controlled by the 10 CFR 50.59 process. As discussed in Generic Letter 88–16, the administrative controls established for the values of cycle specific parameters using the guidance of that letter assure conformance with 10 CFR 50.36. Safety analysis acceptance criteria are not being altered by this amendment. Therefore, operation of the facility in accordance with the 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 this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Frederick J. Hebdon.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: October 30, 1996.

Description of amendment request: The proposed amendment, included as part of the proposed conversion from current Technical Specifications (TS) to improved TS, would relax the required flowrates in core spray, low pressure coolant injection (LPCI), and high pressure coolant injection (HPCI) systems, based on the DAEC loss-of-coolant-accident (LOCA) analysis, using an NRC-approved code, SAFER/GESTR–LOCA.

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 will lower ECCS required flowrates in accordance with accident analysis assumptions. The ECCS subsystems affected by this change are not assumed to be initiators of analyzed events. Therefore, the proposed change does not increase the probability of any accident. The role of these ECCS subsystems is in the mitigation of accident consequences. The proposed change decreases pump flow rate requirements for Core Spray, LPCI and HPCI. The proposed change does not increase the consequences of an accident because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis, uses these reduced pump flow rates as analysis inputs and demonstrates that peak cladding temperatures are maintained within regulatory limits. Therefore, this change will not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change will not involve any physical changes to plant systems, structures, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. As demonstrated in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis, at the reduced flowrates, adequate ECCS capability will still exist to mitigate the consequences of accidents. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not significantly reduce the margin of safety because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis, uses these reduced pump flow rates as analysis inputs. The accident analysis demonstrates that with these reduced ECCS pump flow rates, the peak clad temperature remains below the regulatory limit. Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on 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: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401.

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis, & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

Acting NRC Project Director: Richard P. Savio.

IES Utilities Inc., Docket No. 50–331 Duane Arnold Energy Center, Linn County, Iowa

Date of amendment requests: January 9, 1998.

Description of amendment requests: The proposed amendment would revise the limiting condition for operation for primary containment isolation valves (PCIVs). The revision would allow 72 hours to isolate a failed valve associated with a closed system.

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

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

This change extends the time to isolate single PCIV penetrations from 4 hours to 72 hours. The time allowed to isolate the penetration is not assumed to be an initiator of any analyzed event. The 72 hour period provides the necessary time to perform repairs on a failed containment isolation valve when relying on an intact closed system. Use of a closed system for isolation is directly equivalent to isolating a failed containment isolation valve by use of a single valve. The closed systems are subject to a Type A containment leakage test, are missile protected, and are seismic Category 1 piping. Allowing an additional 68 hours to isolate these penetrations will not significantly increase the consequences of an accident since the intact closed system provides adequate isolation. Also, the consequences of an event occurring during the proposed 72 hour period are the same as those during the current 4 hour period. The 72 hour period is consistent with NRC-approved Traveler TSTF–30, Revision 2. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change extends the time allowed to isolate single PCIV penetrations from 4 hours to 72 hours. The additional 68 hours that the penetrations are not isolated will not create the possibility of a new or different kind of accident. Use of a closed system for isolation is directly equivalent to isolating a failed containment isolation valve by use of a single valve. The closed systems are subject to a Type A containment leakage test, are missile protected, and are seismic Category 1 piping. This change will not physically alter the plant (no new or different type of equipment will be installed). The change in allowed out-of-service-time is consistent with current safety analysis assumptions. Therefore, this 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 a margin of safety.

This change extends the time allowed to isolate single PCIV penetrations from 4 hours to 72 hours. During the additional time allowed, a limiting event would still be assumed to be within the bounds of the safety analysis assuming no single active failure. The 72 hour period is consistent with NRC-approved Traveler TSTF–30, Revision 2. Use of a closed system for isolation is directly equivalent to isolating a failed containment isolation valve by use of a single valve. Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on 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: Cedar Rapids Public Library, 00 First Street, SE., Cedar Rapids, Iowa 52401.

Attorney for licensee: Jack Newman, Al Gutterman, Morgan, Lewis & Brockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Acting Project Director: Richard P. Savio.

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

Date of amendment request: December 11, 1997.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to add a new Limiting Condition for Operation (LCO) for an inoperable engineering safety features (ESF) logic subsystem.

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.

Omaha Public Power District (OPPD) proposes to incorporate a new Limiting Condition for Operation (LCO) into Specification 2.15 which will apply to an engineered safety features (ESF) logic subsystem when the minimum operable channels or minimum degree of redundancy requirements listed in Tables 2–3 and 2–4 are not met. The LCO proposes an allowed outage time (AOT) of 48 hours to restore sufficient channels to operability so as to exceed minimum requirements, or the plant must be placed in hot shutdown within the following 12 hours.

The ESF logic system is a Class 1 protection system designed to satisfy the criteria of IEEE 279, August 1968. Two functionally redundant ESF logic subsystems ''A'' and ''B'' are provided to ensure high reliability and effective in-service testing. These logic subsystems are designed for individual reliability and maximum attainable mutual independence both physically and electrically. Either ESF logic subsystem acting alone can automatically actuate ESF equipment and essential supporting systems.

The design of the ESF logic system is not being altered by this change. The change allows a reasonable time to contact trained personnel and adequately troubleshoot, perform and test repairs on an inoperable ESF logic subsystem. The proposed AOT ensures that repairs are thoroughly planned and accomplished without undue haste. In this situation, the opposite ESF logic subsystem is operable as verified through surveillance testing and capable of providing both automatic and manual ESF equipment actuation.

The proposed AOT is similar to that of LCO 3.3.5, ''Engineered Safety Features Actuation System (ESFAS)

Logic and Manual Trip (Analog),'' of Combustion Engineering Owners Group (CEOG) Standard Technical Specification (STS), Rev. 1, dated April 7, 1995.

Additional administrative revisions are proposed to either support the new LCO (e.g., footnotes in Tables 2–3 & 2–4) or clarify existing information. Therefore, OPPD concludes that the proposed LCO and administrative revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There will be no physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits because of these proposed changes. No changes in operating modes are proposed. The proposed LCO provides a reasonable AOT to troubleshoot, repair, and test an inoperable ESF logic subsystem. The remaining ESF logic subsystem is still operable and capable of both automatic and manual ESF equipment actuation. The remaining changes are administrative in nature and thus none of the proposed changes create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed LCO provides a reasonable AOT to troubleshoot, repair, and test an inoperable ESF logic subsystem. The remaining ESF logic subsystem is still operable as verified by surveillance testing and capable of both automatic and manual ESF equipment actuation. With an inoperable ESF logic subsystem, the ESF logic system would not be single failure proof for a brief period of time. However, it is OPPD's position that making repairs while the plant is at power and stable is preferable to imposing a transient (manual shutdown) on the plant at a time when the ESF logic system is no longer single failure proof. Therefore, OPPD concludes that the proposed LCO and supporting administrative changes do not result in a significant reduction in a margin of safety.

Based on the above considerations, it is OPPD's position that this proposed amendment does not involve significant hazards considerations as defined by 10 CFR 50.92 and the proposed changes will not result in a condition which significantly alters the impact of the Station on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and pursuant to 10 CFR 51.22(b) no environmental assessment need be prepared.

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

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

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: William H. Bateman.

Philadelphia Electric Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of amendment request: January 12, 1998.

Description of amendment request: The Philadelphia Electric Company submitted a Technical Specifications (TS) Change Request, requesting an amendment to the TS (Appendix A) of Operating License No. NPF–39 for Limerick Generating Station (LGS), Unit 1. This proposed change will revise TS Table 4.4.6.1.3–1 to change the withdrawal schedule for the first capsule to be withdrawn from 10 Effective Full Power Years (EFPY) to 15 EFPY.

A revision to TS Surveillance Requirement 4.4.6.1.4 is also proposed. This revision will remove the references to flux wire removal and analysis that was originally required following the first cycle of operation. The referenced flux wires were never located following the first cycle of operation. This TS Surveillance Requirement will be changed to refer to the flux wires that are located within the surveillance capsules, which will be removed and analyzed in accordance with the surveillance capsule removal schedule located in TS Table 4.4.6.1.3–1.

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

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

The proposed changes do not increase the probability of occurrence of an accident previously evaluated in the safety analysis report and do not affect any accident initiators as described in the SAR [Safety Analysis Report]. The changes revise the withdrawal schedule for the reactor vessel material surveillance capsules from 10 Effective Full Power Years (EFPY) to 15 EFPY. The capsules are not an initiator of any previously analyzed accident nor does the withdrawal schedule of the surveillance capsule affect the probability or consequences of any previously analyzed accident.

These changes will not affect the Pressure-Temperature (P–T) limits as given in LGS Technical Specification (TS) Figure 3.4.6.1–1 and UFSAR [Updated Final Safety Analysis Report] Figure 5.3–4. P–T limits are imposed on the reactor coolant system to ensure that adequate safety margins exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P–T limits are related to the RT_{NDT} [reference temperature], as described in ASME Section III, Appendix G. Changes in the fracture toughness properties of reactor pressure vessel (RPV) beltline materials, resulting from neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10 CFR 50 Appendix H. The effect of neutron fluence on the shift in the RT_{NDT} is predicted by methods given in Regulatory Guide 1.99, Rev. 2.

As detailed in Attachment 3 [of the licensee's application dated January 12, 1998], for LGS Unit 1, the combination of low expected RT_{NDT} shift for the plate material due to low predicted fluence and excellent material chemistry, Supplemental Surveillance Program (SSP) data on similar material, and the inherent margin in the P–T curve calculations—with the withdrawal schedule of the first surveillance capsule modified from 10 EFPY to 15 EFPY—will result in a more credible set of surveillance data while ensuring the continued safe operation of LGS Unit 1.

LGS's current P–T limits were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in Regulatory Guide 1.99, Rev. 2, Regulatory Position 1, ''Surveillance Data Not Available.'' Calculation of adjusted reference temperature by these procedures includes a conservative base fluence estimate, power rerate adjustment of a 110% fluence multiplier from startup—instead of a 105% fluence multiplier since 1R06 [Unit 1 refueing outage 6], and a margin term to ensure conservative, upper-bound values are used for the calculation of the P–T limits. Revision of the first capsule withdrawal schedule will not affect the P–T limits because the capsule constitutes one set of credible surveillance data. The curves will continue to be established in accordance with Regulatory Position 1 procedures.

As per Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, Regulatory Position 2, ''Surveillance Data Available,'' the collection of two or more sets of credible surveillance data is necessary to empirically calculate the adjusted reference temperature (ART). Each surveillance capsule constitutes one set of credible surveillance data. This calculated ART can be used to revise the Pressure-Temperature (P–T) curves (Technical Specification Figure 3.4.6.1–1). Without two or more sets of credible data, the ART must be calculated and the P–T curves revised, based upon the calculational methodologies as provided in the Regulatory Guide 1.99, Rev. 2, Regulatory Position 1, ''Surveillance Data Not Available.'' These methodologies use plant specific chemistry and fluence values to determine a calculated shift in RT_{NDT}. A ''margin'' term is then added to obtain conservative, upper-bound values of adjusted reference temperature.

The existing LGS Unit 1 P–T curves are currently valid up to 12 EFPY. With first capsule removal at *either* 10 *or* 15 EFPY, the existing P–T curves will require a revision prior to reaching 12 EFPY based upon the calculational methodologies as contained in the Regulatory Guide 1.99, Rev. 2, Regulatory Position 1, ''Surveillance Data Not Available.'' Therefore, the revision to the first capsule withdrawal schedule results in no impact to the calculational methodologies that will be used for the P–T curve revision that will be necessary to extend the curves beyond 12 EFPY.

The fluence data as determined from the surveillance capsule flux wires at 15 EFPY will provide an accurate indication of neutron fluence. In accordance with Regulatory Guide 1.99, Rev. 2, Regulatory Position 1 methodology, data from these flux wires will permit an adjustment of TS Figure 3.4.6.1–1 in accordance with TS surveillance requirement 4.4.6.1.3, if required, and will meet the requirements of 10 CFR 50 Appendix H and ASTM E–185.

These changes will not affect any plant safety limits or limiting conditions of operation. The proposed changes will not affect reactor pressure vessel performance as they do not involve any physical changes, and LGS P–T limits will remain conservative in accordance with Reg. Guide 1.99, Rev. 2 requirements. The proposed changes will not cause the RPV or interfacing systems to be operated outside of their design or testing limits.

The proposed changes do not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. The proposed changes do not involve any physical changes to equipment important to safety. The potential for RPV failure will be adequately assessed by the proposed withdrawal schedule. In addition, the results from the SSP will provide industry data that bounds the materials used in the LGS Unit 1 reactor pressure vessel until the data from the first LGS Unit 1 capsule is available. The proposed changes provide the same level of confidence in the integrity of the vessel.

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 changes do not create the possibility of a different type of accident than any previously evaluated in the SAR. The proposed changes will revise the withdrawal schedule for the first reactor pressure vessel (RPV) material surveillance capsule from 10 Effective Full Power Years (EFPY) to 15 EFPY. These proposed changes do not involve a physical modification of the design of plant structures, systems or components. The proposed changes will not impact the manner in which the plant is operated, as plant operating and testing procedures will not be affected by the changes. No new accident types or failure modes will be introduced as a result of the proposed changes.

LGS's current Pressure-Temperature (P–T) limits were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in Regulatory Guide 1.99, Rev. 2, Regulatory Position 1, ''Surveillance Data Not Available.'' Calculation of adjusted reference temperature by these procedures includes a conservative base fluence estimate, power rerate adjustment of a 110% fluence multiplier from startup—instead of a 105% fluence multiplier since 1R06, and a margin term to ensure conservative, upper-bound values are used for the calculation of the P–T limits. Revision of the first capsule withdrawal schedule will not affect the P–T limits because the capsule constitutes one set of credible surveillance data. The curves will continue to be established in accordance with Regulatory Position 1 procedures.

The existing LGS Unit 1 P–T curves are currently valid up to 12 EFPY. With first capsule removal at *either* 10 *or* 15 EFPY, the existing P–T curves will require a revision, prior to reaching 12 EFPY, based upon the calculational methodologies as contained in the Regulatory Guide 1.99, Rev. 2, Regulatory Position 1, ''Surveillance Data Not Available.''

Therefore, the Technical Specification (TS) revision to the first capsule withdrawal schedule results in no impact to the calculational methodologies that will be used for the P–T curve revision that will be necessary to extend the curves beyond 12 EFPY.

The fluence data as determined from the surveillance capsule flux wires at 15 EFPY will provide an accurate indication of neutron fluence. In accordance with Regulatory Guide 1.99, Rev. 2, Regulatory Position 1 methodology, data from these flux wires will permit an adjustment of TS Figure 3.4.6.1–1 in accordance with TS Surveillance Requirement 4.4.6.1.3, if required, and will meet the requirements of 10 CFR 50 Appendix H and ASTM E–185.

The potential for reactor pressure vessel (RPV) failure will continue to be adequately assessed by the proposed withdrawal schedule. As detailed in Attachment 3, the combination of the low expected shift for the plate material, SSP data on similar material, and the inherent margin in the P–T curve calculations will result in a credible set of surveillance data, while ensuring the continued safe operation of LGS Unit 1. The proposed changes provide the same level of confidence in the integrity of the RPV.

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

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

The proposed changes to the Technical Specifications (TS) do not reduce the margin of safety as defined in the Bases for any TS. The proposed changes will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed changes do not represent a change in initial conditions, system response time, or in any other parameter affecting the accident analyses supporting the Bases of any TS. The proposed changes do not involve revision of the P–T limits but rather a revision of the withdrawal schedule for the first surveillance capsule. The current P–T limits were established based on the adjusted reference temperatures for vessel beltline materials calculated in accordance with Regulatory Position 1 of Reg. Guide 1.99, Rev. 2. P–T limits will continue to be revised as necessary for changes in adjusted reference temperature due to changes in fluence according to Regulatory Position 1 until two or more credible surveillance data sets become available. When two or more credible surveillance data sets become available, P–T limits will be revised as prescribed by Regulatory Position 2 of Reg. Guide 1.99, Rev. 2 or other NRC approved guidance.

The current P–T limit curves are inherently conservative and provide sufficient margin to ensure the integrity of the reactor pressure vessel. The proposed changes do not adversely affect these curves. The fluence data as determined from the surveillance capsule flux wires at 15 EFPY will provide an accurate indication of neutron fluence.

In accordance with Regulatory Guide 1.99, Rev. 2, Regulatory Position 1 methodology, data from these flux wires will permit an adjustment of TS Figure 3.4.6.1–1 in accordance with TS Surveillance Requirement 4.4.6.1.3, if required, and will meet the requirements of 10 CFR 50 Appendix H and ASTM E–185.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 2, 1997.

Description of amendment request: This proposed Technical Specification (TS) Change Request revises TS Sections 4.0.5, and Bases Sections B 4.0.5 and B 3/4.4.8, for Limerick Generating Station (LGS), Units 1 and 2, pertaining to the surveillance requirement associated with Inservice Inspection (ISI) and Inservice Testing (IST) activities for American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Class 1, 2, and 3 components.

The existing wording in TS Section 4.0.5, and Bases Sections B 4.0.5 and B 3/4.4.8, stipulates that ISI and IST surveillance activities for ASME Code Class 1, 2, and 3 components be conducted in accordance with the requirements of Section XI of the ASME Code as required by 10 CFR 50.55a(g). The proposed changes will revise the applicable TS sections to only make reference to 10 CFR 50.55a, since the current regulations have separated the specific requirements for ISI and IST into sections 50.55a(g) and 50.55a(f), respectively.

The existing wording of TS Section 4.0.5, and Bases Sections B 4.0.5 and B 3/4.4.8, also requires that ISI and IST surveillance activities be conducted in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, except where specific written relief has been granted by the NRC. This wording precludes the immediate implementation of alternative testing in the event that a Code required inspection has been identified as clearly impractical. The proposed TS changes will revise the applicable TS sections to eliminate the requirement that written relief be obtained *prior to* implementation of alternative testing during the initial 120-month inspection interval, and the initial 12 months of subsequent intervals in cases where the Code required inspections have been found to be clearly impractical. NUREG–1482, ''Guidelines for Inservice Testing at Nuclear Power Plants,'' discusses impracticality as being a situation where a test cannot be performed due to limitations in design (which includes prohibitive dose rates), construction, or system configuration.

Furthermore, TS Section 4.0.5b. currently discusses the required frequency of ISI and IST surveillance activities required by the ASME Code. The existing TS address testing frequencies of up to one (1) year. In some cases, the ASME Code requires that testing be performed on a two (2) year frequency. The proposed TS changes will also revise the TS to include a reference for tests that are conducted on a biennial frequency. Inclusion of this reference will permit the application of TS 4.0.2 criteria for ISI and IST surveillance activities. This will permit a 25 percent time extension to be applied to the surveillance frequency, if necessary, in order to allow for consideration of plant operating conditions when scheduling ISI and IST surveillance tests.

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 are administrative in nature and do not make physical modifications or changes to the plant structures, systems, or components (SSC). Plant SSC will continue to function as designed. The proposed TS changes will not alter equipment operational practices or procedures.

In the event that an ASME Section XI Code required inspection or test is found to be impractical due to unforeseen conditions, written relief would still be requested from the NRC in accordance with established procedures. No code required inspection will be eliminated from the ISI or IST Programs until written approval has been granted by the NRC as required [by] 10CFR50.55a. It is anticipated that the only time this provision would be utilized would be in the event that an inspection or test is discovered to be impossible or impractical to perform due to unforeseen or unexpected high radiation conditions, or physical limitations. This change will also clarify the applicability of surveillance intervals to biennial tests or examinations.

The proposed TS changes will remove the inconsistencies between the LGS TS and the requirements of 10CFR50.55a, and will also ensure that the implementation of the LGS ISI and IST Programs are consistent with current NRC guidance as specified in NUREG–1482 and NUREG–1433, Revision 1.

Therefore, the proposed TS changes do not involve a significant 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 changes apply to the administrative requirements for testing of plant systems. No physical modifications to systems or components are involved. No new failure modes which could cause or contribute to the cause of an accident are being introduced.

The proposed TS changes will remove the inconsistencies between the LGS TS and the requirements of 10CFR50.55a, and will also ensure that the implementation of the LGS ISI and IST Programs are consistent with current NRC guidance as specified in NUREG–1482 and NUREG–1433, Revision 1.

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

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

No physical plant modifications or operational procedure changes are being made as a result of the proposed TS changes. The proposed TS changes apply to the ISI and IST Programs' surveillance requirements and do not modify the scope or frequency of these Programs as required by 10 CFR 50.55a. The proposed TS changes will eliminate inconsistencies between current TS wording and the requirements specified in 10CFR50.55a. In addition, the proposed changes are consistent with the guidance stipulated in NUREG–1482 and NUREG–1433, Revision 1. No physical plant modifications or operational procedure changes are being introduced as a result of this proposed TS Change.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: October 8, 1997.

Description of amendment request: This amendment proposes revisions to the actions to be taken in the event multiple control rods are inoperable.

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

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

The number and distribution of inoperable control rods is not a precursor to any accident, therefore the probability of an accident is not affected. The proposed changes assure the assumptions used in evaluation of accidents are satisfied, therefore there will be no increase in the consequences of an accident previously evaluated.

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

Changing the allowable number and distribution of inoperable control rods and the power level at which these limits apply to be consistent with the accident analyses does not create the possibility of a new or different kind of accident.

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

The proposed changes assure the assumptions used in the accident analyses are satisfied, therefore there will be no affect on the margin of safety as a result of these changes.

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

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director.

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 amendment requests: November 6, 1995, as supplemented by letter dated January 9, 1998. The supplemental submittal supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that were published on April 10, 1996 (61 FR 15996).

Description of amendment requests: In the November 6, 1995, letter, the licensee proposed to revise Technical Specification (TS) 3.5.1, ''Safety Injection Tanks,'' to extend, in general, the allowed outage time (AOT) for a single inoperable safety injection tank (SIT) from 1 hour to 24 hours. Additionally, the licensee proposed to extend the SIT AOT from 1 hour to 72 hours if a single SIT becomes inoperable due to malfunctioning SIT water level and/or nitrogen cover pressure instrumentation. The January 9, 1998, letter modifies the original request by adding a new TS 5.5.2.14, ''Configuration Risk Management Program,'' that ensures a proceduralized probabilistic risk assessment-informed process is in place that assesses the overall impact of plant maintenance on plant risk.

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 Safety Injection Tanks (SITs) are passive components in the Emergency Core Cooling System (ECCS). The SITs are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The SITs are designed to mitigate the consequences of Loss of Coolant Accidents (LOCAs). The proposed changes do not affect any of the assumptions used in deterministic LOCA analysis. Therefore, the consequences of accidents previously evaluated do not change.

To fully evaluate the SIT Completion Time extension, Probabilistic Safety Analysis (PSA) methods were utilized. The results of these analyses show no significant increase in core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated.

The proposed change pertaining to SIT inoperability based solely on instrumentation malfunction does not involve a significant increase in the consequences of an accident as evaluated and endorsed by the Nuclear Regulatory Commission (NRC) in

NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements.''

The Configuration Risk Management Program is an Administrative Program that assesses risk based on plant status. Adding the requirement to implement this program for Technical Specification 3.5.1 does not affect the probability or the consequences of an accident.

Therefore, this change does not involve a significant increase in the probability or consequences of any 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.

This proposed change does not change the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrate that the changes are either risk neutral or risk beneficial.

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

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

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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 amendment requests: November 8, 1995, as supplemented by letter dated January 9, 1998. The supplemental submittal supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that were published on April 10, 1996 (61 FR 15996).

Description of amendment requests: In the November 8, 1995, letter, the licensee proposed to revise Technical Specification (TS) 3.5.2, ''ECCS—Operating,'' to extend the allowed outage time from 72 hours to 7 days for a single low pressure safety injection train. The January 9, 1998, letter modifies the original request by adding a new TS 5.5.2.14, ''Configuration Risk Management Program,'' that ensures a proceduralized probabilistic risk assessment-informed process is in place that assesses the overall impact of plant maintenance on plant risk.

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 Low Pressure Safety Injection (LPSI) system is a part of the Emergency Core Cooling System (ECCS) subsystem. Inoperable LPSI components are not considered to be accident initiators. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The LPSI system is primarily designed to mitigate the consequences of a large Loss of Coolant Accident (LOCA). This proposed change does not affect any of the assumptions used in the deterministic LOCA analysis. Therefore, the consequences of accidents previously evaluated do not change.

To fully evaluate the LPSI Completion Time extension, Probabilistic Safety Analysis (PSA) methods were utilized. The results of these analyses show no significant increase in core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated.

The Configuration Risk Management Program is an Administrative Program that assesses risk based on plant status. Adding the requirement to implement this program for Technical Specification 3.5.2 does not affect the probability or the consequences of an accident.

Therefore, this change does not involve a significant increase in the probability or consequences of any 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.

This proposed change does not change the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change does not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes.

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

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

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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 amendment requests: July 29, 1996.

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3.7, ''Plant Systems,'' and TS 4.3, ''Fuel Storage,'' to permit an increase in the licensed storage capacity of the spent fuel pools.

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.

In the course of previous analyses and the analyses required to support the consolidation and storage of spent fuel assemblies generated by the San Onofre Nuclear Generating Station Units 1, 2 and 3 (SONGS 1, 2 and 3), the enveloping scenarios described below have been considered. The limiting event or accident is considered that which produces the greatest radiological dose consequences.

(1) *Design Basis Fuel Handling Accidents.* Postulated fuel handling accidents consider drops of either a spent fuel assembly or a consolidated fuel canister in the spent fuel pool (SFP) or cask pool. In addition to damage to the dropped fuel assembly or consolidated fuel canister, a fuel assembly or consolidated fuel canister seated in the SFP or the cask pool may be impacted by the drop. Alternatively, the dropped assembly or canister may fall over an empty rack cell, or fall onto the pool floor/liner. These various scenarios have been considered.

The reference fuel in the analysis presented below is SONGS 2 and 3 fuel. Due to the longer decay time, lower burnup, and lower operating power of SONGS 1 fuel, the consequences of damage to SONGS 1 fuel are bounded by the consequences of damage to SONGS 2 and 3 fuel.

(a) *Dropped Fuel Assembly.* The limiting and design basis fuel assembly drop event is a 254-inch drop of a vertically-oriented fuel assembly, which has decayed for 72 hours, onto the SFP floor, followed by rotation of the fuel assembly to the horizontal position. The postulated bounding event results in a total of 60 fuel rods failing, which will not change as a result of fuel consolidation.

The probability of a spent fuel assembly drop during movement of spent fuel is slightly increased by fuel consolidation because the candidate fuel assemblies are moved from their individual rack cell location to the cask pool for consolidation. However, this increase in probability is not significant since the process and equipment used to move fuel assemblies will not be changed. Additionally, fuel movement activities will be performed by personnel trained, qualified, and certified in fuel handling operations. Therefore, the increase in probability of a spent fuel assembly drop due to fuel consolidation is not significant.

The SFP water leakage consequences of a fuel assembly drop are bounded by the consequences of a postulated empty spent fuel rack drop. The resulting leakage (approximately 49 gallons per minute) is well within the makeup water supply capability (150 gallons per minute). Additionally, the water loss would be contained within the spent fuel pool leak chase system and would not be released to the soil or the environment.

Spent fuel assemblies will be decayed (subcritical) at least 72 hours prior to being moved and at least 6 months prior to being consolidated. Administrative controls will require that fuel assemblies being moved to and from the consolidation work station, and when in the work station, be separated by more than 12 inches of water from edge to edge to maintain neutronic isolation. Criticality calculations show that with 1800 parts per million (ppm) minimum boron concentration in the SFP water (Technical Specifications limit of 1850 ppm includes 50 ppm measurement uncertainty), a dropped fuel assembly event will not result in fuel criticality.

Without crediting filtration by the fuel handling building (FHB) post-accident cleanup units, the offsite doses which result from this scenario are well within the required limits, i.e., less than 25 percent (%) of the limits imposed by 10 CFR 100. The control room doses meet 10 CFR 50, Appendix A, General Design Criterion (GDC) 19 limits when crediting the control room emergency air cleanup system. Therefore, the consequences of a fuel handling accident remain enveloped by the fuel assembly drop event.

In conclusion, the probability and consequences of a fuel assembly drop event will not be significantly increased by the proposed fuel consolidation activity.

(b) *Dropped Consolidated Fuel Canister.* A dropped consolidated fuel canister event does not involve significantly new failure mechanisms compared with a dropped fuel assembly event. The limiting event in this category is a 74-inch drop of a consolidated fuel canister from the spent fuel handling machine (SFHM) into a rack cell containing a consolidated fuel canister. The structural integrity of the racks would not be impacted and both consolidated fuel canisters would remain intact. However, it is conservatively assumed that all 944 fuel rods within the two canisters (472 rods/canister × 2 canisters) are damaged.

The probability of a consolidated fuel canister drop is not expected to vary significantly from that expected for a fuel assembly drop because the methods and equipment used to move consolidated fuel canisters will not be significantly different from those used for fuel assemblies. Additionally, effective training methods, administrative controls, and equipment design will be developed to minimize the likelihood of dropping a canister during the consolidation process.

The SFP water leakage consequences of a consolidated fuel canister drop are bounded by the consequences of a postulated empty spent fuel rack drop as discussed previously in Item 1.1(a).

The criticality calculations show that, with the required 1800 ppm boron concentration in the SFP and cask pool water, there are no criticality consequences of postulated consolidated fuel canister drops. In all cases, the structural integrity of the racks will be maintained. The portions of the canisters where fuel is contained (above and inclusive of the bottom plate) will maintain their structural integrity in all drop cases.

The offsite doses which result from this scenario are bounded by the fuel assembly drop event discussed previously in Item 1.1(a) (60 failed fuel rods in an assembly which has decayed 72 hours) and are well within (less than 25% of) the limits imposed by 10 CFR 100. The control room doses meet the GDC 19 limits when crediting the control room emergency air cleanup system. Therefore, the consequences of a consolidated fuel canister drop event are enveloped by the limiting fuel assembly drop event.

In conclusion, the probability and consequences of the limiting fuel drop event will not be significantly increased by storing consolidated fuel in canisters.

(2) *Spent Fuel Pool (SFP) Gate Drop.* The limiting case is a SFP gate drop on a fuel assembly. Analysis has shown that only one assembly would be impacted and all 236 rods in the assembly potentially damaged subsequent to a drop of the SFP gate. The radiological consequences are shown to be acceptable (less than 25% of 10 CFR 100 limits).

Current gate lift height restrictions (no more than 30 inches above the racks) will be maintained for fuel consolidation. With these restrictions, fuel in only one rack cell (either a spent fuel assembly with 236 rods or a consolidated fuel canister with 472 rods) would be impacted with all rods in the fuel assembly or canister being potentially damaged.

The probability of a SFP gate drop is not significantly increased by fuel consolidation because the process and equipment used to move the gate will not change and because the gate will be kept open and not moved or removed when fuel is located in the cask pool during consolidation (administrative control).

Despite the additional fuel rods in a consolidated fuel canister (472 rods versus 236 rods in a fuel assembly), the minimum six month decay time allows more than 99.9% of the radioactive gases to decay. Thus, a gate drop that results in a damaged fuel assembly 72 hours after shutdown is more limiting than a gate drop that results in a damaged consolidated fuel canister. With the analysis demonstrating impact of fuel in only one cell, offsite doses remain well within (less than 25% of) the limits of 10 CFR 100 without taking credit for the FHB filters. The control room emergency air cleanup system will maintain control room doses within GDC 19 limits.

Therefore, the probability and consequences of a gate drop will not be significantly increased due to the proposed fuel consolidation activity.

(3) Test Equipment Skid Drop. Current test equipment skid height restrictions (no more than 72 inches above rack cells containing SONGS 2 and 3 fuel assemblies or 30 feet 8 inches above those containing SONGS 1 assemblies) will be maintained after fuel consolidation is implemented. These restrictions will ensure that the potential depth of penetration of test equipment skid into the racks is not sufficient to damage stored fuel.

The probability of a test equipment skid drop is not affected by fuel consolidation because the methods and equipment used to move the skid will not change. In addition, there are no adverse criticality consequences of a test equipment skid drop on a fuel assembly or consolidated fuel canister, since the structural configuration of the fuel or of the impacted storage rack cells is not significantly changed because of the drop impact.

Since no fuel is damaged, the probability and consequences of a test equipment skid drop will not be significantly increased due to the proposed fuel consolidation activity.

(4) Cask Handling Crane Load Drops. The types of loads currently lifted by the cask handling crane include spent fuel casks, transshipment casks, and the crane load block. To support consolidation activities, lifts of the fuel consolidation equipment will also be performed by the cask handling crane. The travel path of the cask handling crane does not extend over spent fuel in the SFP. Administrative controls will prohibit operation of the cask handling crane, including the crane load block, within ten feet of the edge of the cask pool when fuel is present in the cask pool during consolidation. The handling of heavy loads by the cask handling crane is governed by the SONGS heavy loads program which has received Nuclear Regulatory Commission (NRC) approval. The movement of fuel consolidation equipment by the cask handling crane will be evaluated under the heavy loads program. Thus, an accident resulting from cask handling crane load drops into the SFP or onto irradiated fuel in the cask pool is not credible.

It is expected that the consolidation work station in the cask pool will be temporarily removed prior to any spent fuel cask, transshipment cask, or other load lifts/movements over the cask pool. Other than insertion and removal of the consolidation work station, the equipment and procedures used to lift and move cask handling crane loads will be unaffected by fuel consolidation.

Therefore, the probability and consequences of a spent fuel cask or transshipment cask drop are not significantly increased by the proposed fuel consolidation activity.

(5) Mispositioning of a Consolidated Fuel Canister. The probability of mispositioning a consolidated fuel canister is expected to be comparable to that for mispositioning of a spent fuel assembly because the methods and equipment used to move and position consolidated fuel canisters in rack cells will not be significantly different from those used for fuel assemblies. Additionally, fuel movement activities are and will continue to be performed by personnel trained, qualified, and certified in fuel handling operations.

The potential consequences of a mispositioned consolidated fuel canister relate to fuel criticality. The burnup of the fuel stored in the SFP before, during, and after consolidation will conform to the criteria provided in the Technical Specifications. With the minimum required 1800 ppm (1850 ppm plus 50 ppm measurement uncertainty) boron concentration in the SFP and the Region II racks loaded with fuel which meets the burnup criteria of Technical Specification 3.7.18, k-eff remains less than 0.90 for a consolidated fuel canister mispositioned in the Region II racks.

Therefore, the probability and consequences of mispositioning a consolidated fuel canister are not significantly higher than the probability and consequences of mispositioning a fuel assembly.

(6) *Maximum Flow Blockage to Cool Spent Fuel.* Flow blockage to a consolidated fuel canister may be caused by either damage to the canister or loose material in the spent fuel pool or cask pool. Canisters will be inspected prior to being placed in the cask pool (prior to loading with fuel), and if damaged during movement or placement in the spent fuel pool. Additionally, the existing foreign material exclusion control in the spent fuel pool area will be utilized for fuel consolidation. Therefore, the probability of blocking flow to a consolidated fuel canister will not be significantly increased.

The temperature effects of a postulated flow blockage of a consolidated fuel canister were evaluated relative to the anticipated maximum cladding temperature of 700 degrees Fahrenheit (700°F) during reactor full power. Each rack storage cell has large or multiple flow holes to virtually eliminate the possibility that all flow in a cell would be blocked by debris or foreign material. The flow openings in the canisters will be designed to maintain a clear flow area of at least 20% under all postulated blockage conditions. For the postulated 80% flow blockage, the resulting maximum cladding temperature is 233.1°F, which is well below the maximum anticipated cladding temperature of 700°F during reactor full power.

Therefore, the probability and consequences of flow blockage will not be significantly increased by the proposed fuel consolidation activity.

(7) *Loss of Spent Fuel Pool (SFP) Cooling.* The probability of loss of SFP cooling is not affected by fuel consolidation because the existing SFP cooling system will perform its design function without modification.

The overall design basis (maximum abnormal) heat load will be increased due to an increased number of spent fuel elements stored. The cask pool may be used for temporary storage of spent fuel assemblies during consolidation. Loss of cooling flow to the cask pool has not been specifically analyzed. However, because of administrative controls which limit the amount of fuel permitted in the cask pool during consolidation and require the gate between the cask pool and the SFP to be open when fuel is present in the cask pool, this accident scenario is bounded by the SFP boiling case discussed below.

An analysis of loss of SFP cooling has been performed using the design basis consolidated fuel heat load. This analysis shows that, without crediting the FHB filters, the offsite doses will remain well within (less than 25% of) the 10 CFR 100 limits. Since the reactivity will decrease with increasing temperature at 0 ppm boron concentration, there will be no adverse criticality effects. Additionally, the normal makeup sources to the SFP will continue to maintain adequate inventory and flow capacity (150 gallons per minute or gpm) to compensate for evaporative losses due to boiling (<112 gpm maximum). The temperature effects of SFP boiling on the SFP liner plate and concrete structure have been determined to be acceptable.

Therefore, the probability and consequences of a loss of SFP cooling event will not be significantly increased by the proposed fuel consolidation activity.

(8) *Consolidation Work Station Accidents.* Fuel consolidation will require additional fuel handling operations. However, since the fuel handling methods and equipment will not be significantly different from those currently used, consolidation work station accidents will be similar to fuel handling accidents already discussed in this Safety Analysis (dropped fuel assembly, dropped consolidated fuel canister, or other load drops). To avoid a significant increase in the probability of any of these accidents, personnel training methods, equipment design, and administrative controls will be utilized. Administrative controls will require a minimum decay time of six months for spent fuel prior to its movement into the cask pool for consolidation. This restriction ensures that the limiting radiological offsite and control room dose consequences from a work station accident remain bounded by a fuel assembly drop. The results are well within (less than 25% of) 10 CFR 100 and meet GDC 19 dose limits.

Fuel assemblies in the work station shall be separated by more than 12 inches of water from edge to edge to maintain neutronic isolation (administrative control). The total spent fuel which will be permitted in the cask pool at any given time is 553 fuel rods (administrative control). This quantity of fuel is equivalent to two full SONGS 2 or 3 fuel assemblies plus a damaged fuel rod storage canister or basket containing up to 81 fuel rods. A criticality analysis has shown that, in the worst case scenario, at 1800 ppm (Technical Specification limit of 1850 ppm includes 50 ppm measurement uncertainty) boron concentration, k-eff will be below 0.95. Additional administrative controls will be imposed to ensure that a minimum of 400 fuel rods or non-fuel rods will be loaded into a SONGS 2 or SONGS 3 consolidated fuel canister and a minimum of 324 fuel rods or non-fuel rods will be loaded into a SONGS 1 consolidated fuel canister.

The canisters shall be designed for storage of fuel rods within a maximum allowed rod pitch. For canisters not fully loaded, the rod pitch shall be maintained by restraints inserted within the canister to ensure against rod displacement during canister movement (administrative control). These limitations ensure that the k-eff for a loaded consolidated fuel canister will not exceed 0.95 with zero ppm boron concentration, considering worst case pitch between consolidated rods. With 1800 ppm boron concentration in the pool, k-eff will be below 0.88 for the worst case canister pitch between rods. Thus, there are no adverse criticality consequences since the minimum number of rods consolidated in a canister is administratively controlled and SFP and cask pool boron concentration will be maintained at or above 1800 ppm during consolidation.

Therefore, the consequences of a consolidation work station accident are not significantly increased as a result of the proposed fuel consolidation activity.

(9) *Seismic Events.* The probability of occurrence of a seismic event is unaffected by the proposed fuel consolidation activity. The consequences of a design basis earthquake (DBE) have been analyzed, and the fuel consolidation process and consolidated fuel canisters will not affect the ability of the racks to maintain their required design basis function during and after a DBE. The spent fuel racks are designed, and the consolidated fuel canisters will be designed, to Seismic Category I requirements, and the consolidation equipment will be designed to Seismic Category II/I requirements as defined by NRC Regulatory Guide 1.29, Revision 3.

The consolidation process provides the capability to store more spent fuel (up to approximately 2867 fuel assemblies) than previously approved by the NRC (up to 1542 fuel assemblies) in the SFP. The fuel handling building and the SFP and cask pool structures have been evaluated for the increased loading from fully-loaded consolidated fuel canisters and the loads found to be within the design allowables.

Thus, the probability or consequences of a seismic event are not significantly increased by the proposed fuel consolidation activity.

(10) *Consolidated Fuel Canister Stuck in a Spent Fuel Rack.* The probability of a consolidated fuel canister being stuck in a spent fuel rack is not known from experience since fuel consolidation demonstration projects conducted to date have not reported this type of occurrence. However, the canisters will be designed to be handled by the spent fuel handling machine (SFHM), will have the same approximate cross-sectional dimensions as spent fuel assemblies, and similar handling equipment and methods will be used. Therefore, the failure mechanisms are expected to be comparable to those for a stuck fuel assembly. On this basis, the probability of a consolidated fuel canister being stuck in a spent fuel rack is estimated to be comparable to that for a stuck fuel assembly.

The canisters will be designed to accommodate all operational and handling loads. A design requirement will be imposed that the canisters be capable of withstanding the maximum SFHM lift load of 6000 pounds and remain intact with no fuel spillage. This is consistent with the criteria utilized previously during SFP reracking for the spent fuel racks and a jammed fuel assembly. With these design criteria and restrictions, deformation of rack cell geometry would not be sufficient to exceed the criticality acceptance criterion (k-eff≤0.95). Therefore, the consequences of a stuck consolidated fuel canister would be bounded by the consequences of a stuck fuel assembly.

Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated due to the proposed fuel consolidation activity.

(11) *Limiting Component Cooling Water (CCW) System Heat Load Effects on Spent Fuel Pool Cooling.* The maximum calculated heat load for the CCW system occurs during a Loss of Coolant Accident (LOCA). The probability of a LOCA, and therefore the probability of maximum heat load being imposed on the CCW system, is not affected by fuel consolidation. The reason is that spent fuel handling operations in the SFP or the cask pool are not, of themselves, LOCA initiators. For the purposes of assessing the heat load on the CCW system, the LOCA is divided into two phases, ''safety injection'' and ''recirculation.''

During the safety injection phase, the SFP heat load is isolated from the CCW system. During the recirculation phase, CCW system cooling to the SFP may be reestablished manually. The recirculation phase represents the highest design heat load for the CCW system. Considering the limiting consolidated fuel heat load contribution from the SFP (assuming a minimum of 60 days decay of the most recent half-core discharged into the SFP), the CCW system has adequate capacity to still remove its design basis heat load.

Therefore, the probability or consequences of a limiting design basis heat load event on the CCW system are not significantly increased by the proposed fuel consolidation activity.

Therefore, operation of the facility in accordance with this proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or

different kind of accident from any accident previously evaluated.

The proposed change will allow the consolidation of San Onofre Units 1, 2 and 3 spent fuel in canisters and the storage of these canisters along with fuel assemblies in the Units 2 and 3 spent fuel pools. Fuel consolidation is similar in nature to fuel reconstitution within a fuel assembly since individual rods are manipulated in both processes. Accidents involving consolidated fuel canisters are similar in nature to fuel assembly handling accidents since both use similar fuel handling processes and equipment. Administrative controls will be instituted to provide assurance that postulated events involving consolidated fuel will be enveloped by the spectrum of design basis fuel handling accidents. Furthermore, heavy load drops during spent fuel handling operations are accidents that have been previously evaluated. Additional evaluations have been performed to demonstrate that when the minimum boron concentration requirements of the Technical Specifications have been met, the criticality criterion is satisfied for all postulated accidents.

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

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

The issue of ''margin of safety,'' when applied to spent fuel consolidation and storage, includes the following areas:
(1) Nuclear criticality,
(2) Thermal-hydraulics,
(3) Mechanical, material and structural aspects, and
(4) Offsite doses.

These four areas are addressed below.

(1) *Nuclear Criticality.* The margin of safety that has been established for nuclear criticality is that, including all uncertainties, there is a 95% probability at a 95% confidence level that the effective neutron multiplication factor (k-eff) in spent fuel pools shall be less than or equal to 0.95, under all normal and postulated accident conditions. This margin of safety has been adhered to in the criticality analyses for fuel consolidation and the storage of consolidated fuel canisters.

Criticality of fuel assemblies and consolidated fuel canisters in fuel storage racks is prevented by the rack design which precludes interactions between two fuel assemblies or two consolidated fuel canisters or between a fuel assembly and a consolidated fuel canister. This is accomplished by fixing the minimum separation between storage cells containing fuel assemblies or consolidated fuel canisters, using Boraflex, a neutron absorbing material, and utilizing strict administrative controls.

During the consolidation process, fuel rods which cannot be consolidated will be placed in a damaged fuel rod canister or basket. Fuel assemblies, consolidated fuel canisters, and damaged fuel rod canisters or baskets moving to and from the consolidation work station or present in the work station shall be separated by more than 12 inches of water, measured edge to edge, to ensure that they are neutronically isolated (administrative control). The total spent fuel which will be permitted in the cask pool at any give time is 553 fuel rods (administrative control). This quantity of fuel is equivalent to two full SONGS 2 or 3 spent fuel assemblies plus 81 fuel rods in a damaged fuel rod canister or basket. Additionally, the rod pitch inside partially loaded canisters shall be maintained by restraints inserted within the canister to ensure against rod displacement during canister movement (administrative control).

The analytical methods utilized in the criticality analyses conform with American National Standards Institute (ANSI) Standard N18.2–1973, ''Nuclear Safety Criteria for the Design of Stationary Pressurizer Water Reactor Plants,'' Section 5.7, Fuel Handling Systems; ANSI Standard 57.2–1983, ''Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations,'' Section 6.4.2; ANSI Standard N16.9–1975, ''Validation of Calculational Methods for Nuclear Criticality Safety;'' NRC Standard Review Plan (NUREG–0800), Section 9.1.2, ''Spent Fuel Storage''; and the NRC guidance, ''OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,'' (April 1978), as modified (January 1979).

The criticality analyses performed for normal conditions assume zero boron concentration in the SFP water and worst-case fuel enrichments and burnups. Most credible accident conditions will not result in an increase in k-eff of the spent fuel racks. However, accidents, such as a heavy load drop, misloading a consolidated fuel canister or dropping a fuel assembly, can be postulated to increase reactivity. For these accident conditions, the double contingency principle of ANSI N16.1–1975 is applied. This principle states that it is not required to assume two unlikely, independent events to ensure protection against a criticality accident. Therefore, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since the absence of boron would be the second unlikely event.

Worst case accident analyses have been performed that show that 1800 ppm of soluble boron will maintain the spent fuel pool and cask storage pool k-eff less than 0.95, including uncertainties, at the required 95%/95% probability/confidence level.

(2) *Thermal-Hydraulics.* The relevant thermal-hydraulics considerations for determining if there is significant reduction in a margin of safety are: (1) maximum fuel temperature, and (2) increase in temperature of the water in the pool, and (3) increase in heat load rejection to the environment.

Similar to the criticality analysis, the SFP decay heat load calculation assumes worst-case fuel loading, enrichment, and burnup. The calculation uses the same methodology as that used for the original decay heat analysis. Standard Review Plan (SRP) Section 9.1.3 criteria for maximum normal and maximum abnormal heat load conditions were used in this evaluation.

The effect of the increased heat load has been evaluated and it has been shown that, under the SRP maximum normal heat load, the existing spent fuel pool cooling system will maintain the bulk pool water temperature below 145°F. This value considers a single active failure of one spent fuel pool cooling system pump, coincident with a loss of offsite power, and is consistent with Standard Review Plan, Section 9.1.3.III.1.d. The 145°F temperature represents a small increase in the currently approved SFP temperature of 140°F. However, this temperature limit was very conservatively calculated, considering only heat losses through the spent fuel pool heat exchangers, and conservatively neglecting losses through evaporation to the spent fuel pool area, as well as conduction to the fuel handling building structure mass. This increase in spent fuel pool temperature does not represent a significant reduction in the margin of safety, since the affected portions of the spent fuel pool cooling system and other important to safety equipment in the fuel handling building are qualified for this slightly higher temperature and will still perform the necessary safety functions when required.

A thermal-hydraulic analysis has been performed which shows that the maximum local water temperatures along the fuel channels will remain below the nucleate boiling condition values, even with the maximum postulated flow blockage (80%) of the consolidated fuel canisters. The maximum calculated fuel cladding temperature for the design basis condition is 233.1°F, which is well below the anticipated maximum cladding temperature of 700°F during full power operation of the reactor.

SONGS 2 and 3 conduct refueling by offloading either half the core or the full core. The full core offload refueling provides the greater of the two heat loads. Therefore, in addition to the SRP criteria, the heat load during refueling operations was also evaluated. For this case the heat load was evaluated assuming a two year refueling cycle, the spent fuel pool completely filled with consolidated fuel (except for the last core offload), and the full core offloaded at 150 hours of decay. Under these conditions, a single SFP cooling pump with two heat exchangers will maintain the SFP temperature below 160°F, assuming the component cooling water temperature is 88°F and the ocean water temperature is 76°F. Thus, the SFP cooling system meets the single active failure criterion for the maximum refueling heat load condition.

With the postulated SRP maximum abnormal heat load, the bulk pool temperature will reach a maximum of 160°F with two pumps and two heat exchangers in operation. This maximum temperature is well below the SRP maximum temperature limit of 212°F. Also, according to the SRP guidance, a single active failure need not be considered for the maximum abnormal heat load case.

The shutdown cooling system (SDCS), if available, can be used as an alternate heat dissipation path for cooling the SFP. The SDCS has been evaluated for the maximum normal and maximum abnormal heat loads and it has been determined that the system and interconnecting ties are adequate to maintain the SFP temperature below 145°F for the maximum normal heat load and below 160°F for the maximum abnormal heat load. Since the maximum abnormal heat load bounds the maximum refueling heat load, there is no need to evaluate the SDCS for the maximum refueling heat load. For the maximum refueling heat load, the SDCS does not meet the single failure criterion for SFP cooling; however, the use of the SDCS for SFP cooling during Modes 5 and 6 of plant operation has previously been evaluated and considered acceptable by the NRC.

The heat load rejection to the environment will only increase by approximately 0.03%.

Thus, there is no significant reduction in a margin of safety, as determined by thermal-hydraulics considerations.

(3) *Mechanical, material, and structural aspects.* The main safety function of the spent fuel pool and the storage racks is to maintain the spent fuel assemblies and consolidated fuel canisters in a safe configuration through normal and/or abnormal loadings. Abnormal loads include an earthquake, impact due to a cask drop, drop of a spent fuel assembly or consolidated fuel canister, or drop of a heavy load including a spent fuel pool gate. The mechanical, material, and structural design of the consolidation work station and consolidated fuel canisters will be in accordance with the applicable portions of the ''NRC OT Position of Review and Acceptance of Spent Fuel Storage and Handling Applications'' and other applicable NRC guidance and industry codes. The canisters will be designed to Seismic Category I requirements, and the consolidation equipment will be analyzed and either restrained or anchored as appropriate to meet Seismic Category II/I requirements as defined by NRC Regulatory Guide 1.29, Revision 3. The consolidation work station and consolidated fuel canister materials will be compatible with the spent fuel rods and spent fuel assemblies, and the spent fuel pool water chemistry. Therefore, margins of safety relative to mechanical, material, and structural aspects of the proposed fuel consolidation activities will not be significantly reduced.

(4) *Offsite and Control Room Doses.* The offsite and control room dose consequences of accidents involving consolidated fuel canisters or fuel consolidation activities were evaluated. To determine the radiological consequences, all credible accidents related to fuel consolidation activities were considered. The analyses assume that spent fuel has decayed a minimum of 6 months prior to commencing the consolidation process.

The limiting accident for fuel consolidation is a 74-inch drop of a consolidated fuel canister from the Spent Fuel Handling Machine (SFHM) onto a rack cell containing a consolidated fuel canister. Although both consolidated fuel canisters would remain intact, it is conservatively assumed that all 944 fuel rods within the two canisters (472 rods/canister × 2 canisters) are damaged. The resultant release of radioactivity, after escaping from the spent fuel pool, is exhausted from the fuel handling building (FHB) over a two-hour period; no credit for FHB isolation system or FHB filters was taken.

The results demonstrate that, with a minimum decay time of 6 months and no credit taken for isolation or filtration, the radiological consequences of the worst case consolidated fuel accident would not result in releases that would exceed 25% of the 10 CFR 100 limits. The results also demonstrate that the control room doses would meet the 10 CFR 50, Appendix A, GDC 19 limits when crediting the control room emergency air cleanup system.

Therefore, operation of the facility according to this 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 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: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: T.E. Oubre, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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 amendment requests: January 24, 1997.

Description of amendment requests: The licensee proposes to revise Surveillance Requirement 3.8.1.9 to Technical Specification 3.8.1, ''AC Sources—Operating.'' This change will revise the surveillance requirement to more accurately reflect safety analysis conditions.

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

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

The proposed change would revise Surveillance Requirement (SR) 3.8.1.9 to more clearly reflect test conditions and be in greater agreement with NUREG 1432.

The Voltage and Frequency limits are made tighter, to accurately reflect plant design requirements. Discussion regarding reactive power loading is eliminated from the SR, consistent with the wording of NUREG 1432, Rev. 1, and added to the Bases.

Operation of the facility would remain unchanged as a result of the proposed changes and no assumptions or results of any accident analyses are affected. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change would revise Surveillance Requirement (SR) 3.8.1.9 to more clearly reflect test conditions and be in greater agreement with NUREG 1432.

Operation of the facility would remain unchanged as a result of the proposed change. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change would revise Surveillance Requirement (SR) 3.8.1.9 to more clearly reflect test conditions and be in greater agreement with NUREG 1432. The Voltage and Frequency limits are made more restrictive, to accurately reflect the assumptions made in the SONGS accident analysis. Consequently, no reduction in any margin to safety exists.

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 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: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: T.E. Oubre, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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

Date of amendments request: December 31, 1997.

Description of amendments request: The proposed amendments would revise the Technical Specifications to change the nuclear instrumentation system intermediate range neutron flux reactor trip setpoint and allowable value.

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 in Intermediate Range reactor trip setpoint from 25% RTP [rated thermal power] to 35% RTP, the associated allowable value change, and the deletion of the redundant references to the IR [intermediate range] high flux and PR [power range] high flux low setpoints do not involve a significant increase in the probability or consequences of an accident previously evaluated in the Farley FSAR [Final Safety Analysis Report]. The IR reactor trip neither causes any accident nor provides primary protection for any accident in the Farley FSAR. No new accident initiators have been identified because of this proposed revision. No new performance requirements for any system that is used to mitigate dose consequences have been imposed by this proposed change. No input assumption to any dose consequence calculation is affected by this proposed change. All previously reported dose consequences remain bounding. Therefore, the radiological consequences to the public resulting from any accident previously evaluated in the FSAR have not significantly increased.

2. The proposed Technical Specifications change to the IR reactor trip setpoint, associated allowable value change, and the deletion of the redundant references to the IR high flux and PR high flux low setpoints do not create the possibility of a new or different kind of accident from any previously evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the increase in IR setpoint from 25% RTP to 35% RTP. No new challenges to the safety-related Reactor Trip System have been identified. The NIS [nuclear instrument system] hardware has not been modified, and Farley will continue to perform periodic IR channel calibration and surveillance in accordance with Technical Specifications. All previously identified accident scenarios remain bounding since the IR trip setpoint provides no primary accident protection. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed increase in the IR reactor trip setpoint from 25% RTP to 35% RTP, the associated allowable value change, and the deletion of the redundant references to the IR high flux and PR high flux low setpoints do not involve a significant reduction in the margin of safety. All previously established acceptance limits continue to be met for all events, since the IR trip does not provide any primary protective action for any accident scenario. Changing the IR setpoint and allowable value will not invalidate its backup function. There are no physical modifications required for the protection system. This change will not affect the operation of any other safety-related equipment. Farley-specific setpoint uncertainty calculations support the setpoint change. Since all acceptance limits continue to be met, there is no significant reduction in the margin of safety.

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

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: January 22, 1998. The application supersedes, in its entirety, the application dated September 13, 1996.

Description of amendment request: The proposed application would change the Vogtle Electric Generating Plant (VEGP) Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' as follows: (1) The completion time for restoration of one required offsite circuit would be increased from 6 to 14 days from discovery of failure to meet the Limiting Condition for Operation (LCO); (2) a new required action B.2 would be added along with the existing Condition B required actions for one Diesel Generator (DG) inoperable, to verify the availability of the Standby Auxiliary

Transformer (SAT) within 1 hour and once per 12 hours thereafter, and restore the DG to operable status within 14 days from discovery of failure to meet the LCO; (3) a new required action B.5.1 would be added to verify that the combustion turbine electrical power generation capability of Plant Wilson is functional and sufficiently reliable to provide assurance of black-start generation capability within 72 hours of entry into Condition B or within 72 hours prior to entry into Condition B; (4) a new required action B.5.2 would be added for utilization when the combined combustion turbine generator (CTG) enhanced black start reliability falls below the required criteria. This condition allows the option to start or run at least one of the CTGs at Plant Wilson within 72 hours of entry to Condition B, or prior to entry into Condition B for preplanned maintenance; (5) a new condition C is being added for when one DG is inoperable and the required actions and completion times of B.2 are not met, i.e. the SAT is not verified to be available or becomes unavailable as an offsite source, or the required actions and completion times of B.5 associated with CTG operation and/or reliability are not met, then restore the DG to operable status within 72 hours; and (6) other changes associated with TS 3.8.1 conditions, required actions, or completion times are only the result of re-numbering due to the addition of the new condition and required actions of the DG extended Allowable Out-of-Service Time (AOT) and do not reflect a change to operating requirements.

In addition, a new TS 5.5.18, ''Configuration Risk Management Program (CRMP),'' would be added to the Administrative section of the TS. This section discusses the program description and use.

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?

No. The DGs are used to support mitigation of the consequences of an accident; however, they are not considered the initiator of any previously analyzed accident. The use of the SAT as an additional offsite power source coupled with the black start generation capability of Plant Wilson and the use of a configuration risk management program will more than compensate for the risk introduced by the extended DG Completion Times. As such, the extension of the DG Completion Times will not significantly increase the probability or consequences of any accident previously evaluated.

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

No. The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. This proposed TS only affects the length of the allowed outage time for DGs and does not change the DG testing or maintenance requirements. The proposed TS still requires the DGs to be maintained Operable to the same standard as before. The use of the SAT as an additional offsite power source coupled with the black start generation capability of Plant Wilson and the use of a configuration risk management program has been shown to provide more than adequate compensation for the potential risk of the extended DG Completion times. The proposed change in DG completion times in conjunction with the added availability of the SAT, continue to provide adequate assurance of the capability to provide power to the ESF [Engineered Safety Features] buses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: December 30, 1997.

Description of amendment request: The proposed amendment would change Table 3.5–1 and associated notes. The changes would remove a potential non-conservative operating configuration for the Residual Heat Removal Service Water (RHRSW) System pumps that could result in a loss of two pumps following a single failure of diesel-generator A or B thereby reducing the number of pumps available to less than the number required by the Final Safety Analysis Report. The changes also would allow (for units with fuel loaded) reducing the minimum-required number of RHRSW pumps by one pump for each unit that has been in cold shutdown for more than 24 hours. The associated Basis 3.5 also would be changed to reflect these changes.

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

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the proposed changes do not involve any plant structures, systems, or components that are initiators of any accident previously evaluated, and the changes do not decrease the capability of the RHRSW system to transfer reactor core and emergency equipment heat loads to the ultimate heat sink.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because there are no changes to plant structures, systems, or components, and the changes do not affect the manner by which the facility is operated. The proposed changes are consistent with the Final Safety Analysis Report analysis for the design basis accident.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes do not affect the manner by which the facility is operated or involve equipment or features which affect the operational characteristics of the facility. The proposed amendment would increase the diversity of power supplies associated with the residual heat removal cooling function thereby improving conformance to the single failure criterion.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

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

NRC Project Director: Frederick J. Hebdon.

Vermont Yankee Nuclear Power Corporation, Docket Nos. 50–271, Vermont Yankee Nuclear Power Station, Windham County, Vermont

Date of amendment request: December 11, 1997.

Description of amendment request: The proposed amendment would revise the safety limit minimum critical power ratio (SLMCPR) values for Cycle 20 operation. The specific changes are:

(1) Page 6, Technical Specification 1.1A. replace the cycle number (19) to (20) and the SLMCPR for Cycle 19 (1.10) with that for Cycle 20 (1.11).

(2) Page 6, Technical Specification 1.1A. replace the SLMCPR for Cycle 19 single loop operation (1.12) with the Cycle 20 value (1.13).

Calculations for Vermont Yankee Nuclear Power Station (VYNPC) by General Electric Company have determined that the current SLMCPR values for single and dual loop operation contained in the Technical Specifications (1.10 and 1.12) are not applicable to the upcoming fuel cycle (Cycle 20) due to core loading design and fuel type changes. The Cycle 20 values are 1.11 and 1.13.

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 basis of the Safety Limit MCPR is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR preserves the existing margin to transition boiling and the probability of fuel damage is not increased. The derivation of the revised SLMCPR for Vermont Yankee Cycle 20 for incorporation into the Technical Specifications, and its use to determine cycle-specific thermal limits, have been performed using NRC approved methods. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

Based on the above, VYNPC has concluded that the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes result only from a specific analysis for the Vermont Yankee Cycle 20 core reload design. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes.

Based on the above, VYNPC has concluded that the proposed change will not create the possibility of a new or different kind of accident from those previously evaluated.

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

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

As a result, VYNPC has concluded that the proposed change will not result in 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

NRC Project Director: Ronald Eaton.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

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

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

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

Date of application for amendment: December 5, 1997.

Brief description of amendment: Revisions to the Crystal River Unit 3 design basis relating to starting logic of reactor building fan coolers.

Date of publication of individual notice in the **Federal Register**: January 15, 1998 (63 FR 2423).

Expiration date of individual notice: February 17, 1998

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal River, Florida 34428.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: October 2, 1997.

Brief description of amendment: The proposed change would revise the Updated Final Safety Analysis Report to revise the credit assumed for iodine decontamination by the spent fuel pool water during a postulated fuel handling accident.

Date of issuance: January 27, 1998.

Effective date: January 27, 1998.

Amendment No.: 177.

Facility Operating License No. DPR–23: Amendment authorizes changes to the facilitiy's Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61838). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

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

Date of application for amendments: May 21, 1996, as supplemented on November 18, 1997, December 3, 1997, January 8, 1998 and January 13, 1998.

Brief description of amendments: The amendments relocate the reactor coolant system pressure and temperature limits for heatup, cooldown, low-temperature operation and hydrostatic testing, and the low-temperature overpresssure protection (LTOP) system setpoint curves into a Pressure Temperature Limits Report (PTLR).

Date of issuance: January 23, 1998.

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

Amendment Nos.: 98, 98, 89, 89.

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

Date of initial notice in **Federal Register:** December 18, 1997 (62 FR 66394). The January 8, 1998 and January 13, 1998, submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 23, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 30, 1997, as supplemented by letter dated December 9, 1997. Additional information was submitted in ComEd's letters of May 23, 1997, August 8, 1997 and January 7, 1998.

Brief description of amendments: The amendments revise the technical specifications and associated bases related to the primary containment pressure and reactor coolant system volume. The changes resulted from the replacement of the steam generators at Byron, Unit 1 and Braidwood, Unit 1.

Date of issuance: January 22, 1998.

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

Amendment Nos.: 97, 97, 88 and 88.

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

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19826) and December 19, 1997 (62 FR 66699).

The May 23, 1997, August 8, 1997, December 9, 1997 and January 7, 1998, letters provided additional information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 22, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 18, 1997, as supplemented by letter dated September 22, 1997.

Brief description of amendments: The amendments change the Technical Specification requirements for steam generator water level to support steam generator replacement at Byron, Unit 1, and Braidwood, Unit 1.

Date of issuance: January 15, 1998.

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

Amendment Nos.: 96, 96, 87 and 87.

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

Date of initial notice in **Federal Register**: March 12, 1997 (62 FR 11491). The September 22, 1997, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 15, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 17, 1997, as supplemented November 26, 1997, and January 9, 1998.

Brief description of amendments: The amendments revise the technical specifications to update the containment vessel structural integrity surveillance requirements to meet the provisions of a recent revision to 10 CFR 50.55a, and to relocate details of the surveillance requirements to a licensee-controlled program.

Date of issuance: January 29, 1998.

Effective date: Effective immediately and shall be implemented within 60 days.

Amendment Nos.: 99, 99, 90 and 90.

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

Date of initial notice in **Federal Register**: December 19, 1997 (62 FR 66697). The November 26, 1997, and January 9, 1998, letters provided additional clarifying information that did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 8, 1997, as supplemented on January 6, 1998.

Brief description of amendments: The amendments revise Technical Specification (TS) 4.5.2.b.3 and the associated Bases to bring the Byron, Unit 1, and Braidwood, Unit 1, requirements into conformance with the Unit 2 requirements that were approved on August 13, 1997. The revision adds a requirement to the Unit 1 TS Surveillance Requirements for verifying that the Chemical and Volume Control (CV) System is full of water every 31 days; to include ultrasonically examining the piping at the CV206 valve for Byron, Unit 1 (CV207 valve for Braidwood, Unit 1), if the train B CV pump is idle. The revision also removes the condition that the Unit 1 requirements will be applicable only until the end of the current cycle (Unit 1-Cycle 8 for Byron, and Unit 1-Cycle 7 for Braidwood). The amendments affect Unit 2 only in that the units share common TS.

As an administrative action by the NRC that only involves the format of the licenses and does not authorize any activities outside the scope of the applications, the NRC has amended the Byron and Braidwood operating licenses to include an Appendix C, ''Additional Conditions,'' and added a license condition associated with the proposed TS changes.

Date of issuance: January 30, 1998.

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

Amendment Nos.: 100, 100, 91 and 91.

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

Date of initial notice in **Federal Register**: November 5, 1997 (62 FR 59914). The January 6, 1998, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 30, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 12, 1997.

Brief description of amendments: The amendments revise the LaSalle County Station Technical Specifications by removing Surveillance Requirement 4.7.1.3.c which requires that every 18 months all areas within the lake screenhouse be inspected to ensure that sediment has not been deposited to a depth greater than 1 foot.

Date of issuance: January 23, 1998.

Effective date: Immediately, to be implemented prior to restart from L1F35 for Unit 1 and prior to restart from L2RO7 for Unit 2.

Amendment Nos.: 122 and 107.

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

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54870).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 23, 1998.

No significant hazards consideration comments received: No.

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

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 application for amendments: September 11, 1997.

Brief description of amendments: These amendments relocate the reactor trip system and engineered safety feature actuation system reponse times from Technical Specification (TS) Tables 3.3–2 and 3.3–5 to Section 3 of the Beaver Valley Power Station, Unit Nos. 1 and 2 Licensing Requirements Manual (LRM) in accordance with the guidance provided in NRC Generic Letter 93–08. Neither the response time limits nor the surveillance requirements for performing response time testing are altered by these amendments. Any future changes to the LRM will be controlled in accordance with the requirements of 10 CFR 50.59. These amendments also make several editorial changes in TSs 3.3.1.1 and 3.3.1.2, as well as making conforming changes to the Bases for these TSs.

Date of issuance: January 20, 1998.

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

Amendment Nos.: 210 and 88.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications and Appendices C (Unit No. 1) and D (Unit No. 2) of the Licenses.

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54871).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 20, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 14, 1997, supplemented August 4, September 2, 17, 25, November 5, 15, 19, 21, December 3, 5, 11, 24, 1997, January 15, and 22, 1998.

Brief description of amendment: Changes to Technical Specification (TS) relating to small break loss of coolant accident mitigation, emergency diesel generator (EDG) upgrade and EDG load rejection test and steady state loads.

Date of issuance: January 24, 1998.

Effective date: January 24, 1998.

Amendment No.: 163.

Facility Operating License No. DPR–72: Amendment revised the TS.

Date of initial notice in **Federal Register**: October 8, 1997 (62 FR 52581). The letters dated August 4, September 2, 17, 25, November 5, 15, 19, 21, December 3, 5, 11, 24, 1997, and January 15, and 22, 1998, provided clarifying 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 January 24, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 1, and 13, 1997 and January 19, 1998.

Brief description of amendment: Revise License Condition 2.C.(5) to delete the requirement relating to installation and testing of flow indicators in the emergency core cooling system to provide indication of 40 gallons per minute flow for boron dilution.

Date of issuance: January 27, 1998.

Effective date: January 27, 1998.

Amendment No.: 164.

Facility Operating License No. DPR–72: Amendment revises License Condition 2.C.(5) and adds a new License Condition 2.C.11.

Date of initial notice in **Federal Register**: November 12, 1997 (62 FR 60733). Letters dated December 1 and 13, 1997 and January 19, 1998 provided supplemental information which did not affect the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1998.

No significant hazards consideration comments received: No.

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

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: November 14, 1997.

Brief description of amendment: The amendment changes Technical Specification 4.5.2.d.1 to clarify the wording and increase the setpoint for the open pressure interlock.

Date of issuance: January 23, 1998.

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

Amendment No.: 156.

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

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66138).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: January 9, 1995, as supplemented by letters dated October 17, 1996, and January 26, 1998.

Brief description of amendment: The amendment revises the technical specifications by deleting toxic gas monitoring requirements for all chemicals except ammonia. The monitoring requirements for ammonia will remain in the technical specifications.

Date of issuance: January 26, 1998.

Effective date: January 26, 1998.

Amendment No.: 183.

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

Date of initial notice in **Federal Register**: March 1, 1995 (60 FR 11137).

The October 17, 1996, and January 26, 1998, supplemental letters provided additional clarifying information and did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 26, 1998.

No significant hazards consideration comments received: No.

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

Philadelphia Electric Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: October 24, 1997.

Brief description of amendment: This amendment changes Sections 3.1.3.6 and 4.1.3.6 of the Unit 1 Technical Specifications to allow operation of control rod 50–27, uncoupled from its driver, for the remainder of Cycle 7. The amendment specifies conditions under which control rod 50–27 may be operated and modifies existing surveillance requirements to verify control rod position by use of neutron instrumentation.

Date of issuance: January 16, 1998.

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

Amendment No.: 124.

Facility Operating License No. NPF–39: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61844).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 8, 1997, as supplemented November 3, 1997.

Brief description of amendment: The requested amendment modifies the f(ΔI) function. The f(delta I) function is defined in the TS as a function of the indicated difference between the top and bottom detectors of the power range nuclear ion chambers. This function is used in the calculation of the overtemperature delta T (OTdelta T) reactor trip.

Date of issuance: January 26, 1998.

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

Amendment No.: 177.

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

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54876). The November 3, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 26, 1998.

No significant hazards consideration comments received: No.

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

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: S. Singh Bajwa.

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

Date of application for amendment: December 11, 1997.

Brief description of amendment: The amendment provides a one-time change to the Technical Specifications to allow purging of the containment during Modes 3 (Hot Standy) and 4 (Hot Shutdown) upon the return to power from the current refueling outage (1R13).

Date of issuance: January 29, 1998.

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

Amendment No.: 206.

Facility Operating License No. DPR–70: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 18, 1997 (62 FR 66397).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 29, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 21, 1997.

Brief description of amendments: These amendments revise the Technical Specifications to extend the Modes from 1 and 2 that the Reactor Trip System Power Range Nuclear Instrumentation—low setpoint is to be operable to Modes 1, 2, and 3, when the reactor trip breakers are in the closed position and the control drive system is capable of rod withdrawl.

Date of issuance: January 29, 1998.

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

Amendment Nos.: 205 and 187.

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

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 68146).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: June 30, 1997, as supplemented September 25, 1997.

Brief Description of amendments: The amendments change the Technical Specifications to incorporate requirements necessary to change the basis for prevention of criticality in the fuel storage pool. The change eliminates the credit for Boraflex as a neutron absorbing material in the fuel storage pool criticality analysis.

Date of issuance: January 23, 1998.

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

Amendment Nos.: Unit 1–133; Unit 2–125.

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

Date of initial notice in **Federal Register**: August 27, 1997 (62 FR 45464).

The staff found that the supplement did not change the conclusions of the proposed no significant hazards consideration; therefore, renotification of the Commission's proposed determination of no significant hazards consideration was not necessary.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 23, 1998.

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.

Dated at Rockville, Maryland, this 4th day of February 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–3269 Filed 2–10–98; 8:45 am]

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OFFICE OF MANAGEMENT AND BUDGET

Budget Rescissions and Deferrals

To the Congress of the United States

In accordance with the Congressional Budget and Impoundment Control Act of 1974, I herewith report eight new deferrals of budgetary resources, totaling $4.8 billion.

These deferrals affect programs of the Department of State, the Social Security Administration, and International Security Assistance.

William J. Clinton

The White House,
February 3, 1998

BILLING CODE 3110–01–P