that such changes would not adversely affect plant safety. The proposed changes have no adverse effect on the probability of any accident. As a result, there is no increase in individual or cumulative radiation exposure.

The environmental impacts of transportation resulting from the use of higher enrichment and extended irradiation are discussed in the staff assessment entitled ''NRC Assessment of the Environmental Effects of Transportation Resulting from Extended Fuel Enrichment and Irradiation.'' This assessment was published in the Federal Register on August 11, 1988 (53 FR 30355), as corrected on August 24, 1988 (53 FR 32322), in connection with the Shearon Harris Nuclear Power Plant, Unit I: Environmental Assessment and Finding of No Significant Impact. As indicated therein, the environmental cost contribution of an increase in fuel enrichment of up to 5 weight percent U–235 and irradiation limits of up to 60 Gigawatt Days per Metric Ton (GWD/MT) are either unchanged, or may in fact be reduced from those summarized in Table S–4 as set forth in 10 CFR 51.52(c). These findings are applicable to the proposed amendment for D.C. Cook Units 1 and 2. Accordingly, the Commission concludes that this proposed action would result in no significant radiological environmental impact.

With regard to potential nonradiological impacts, the proposed changes involve systems located within the restricted area as defined in 10 CFR Part 20. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, the Commission concludes that there are no significant nonradiological environmental impacts associated with the proposed amendment.

The 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 this action was published in the Federal Register on April 24, 1996 (61 FR 18172).

Alternative to the Proposed Action

Since the Commission concluded that there are no significant environmental effects that would result from the proposed action, any alternative with equal or greater environmental impacts need not be evaluated.

The principal alternative would be to deny the requested amendment. This would not reduce environmental impacts of plant operation and would result in reduced operational flexibility.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for D.C. Cook, Units 1 and 2, dated August 1973.

Agencies and Persons Contacted

In accordance with its stated policy, on December 20, 1996, the Commission consulted with the Michigan State official, Mr. Dennis Hahn of the Michigan Department of Public Health, Nuclear Facilities and Environmental Monitoring, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to this action, see the application for license amendment dated February 26, 1996. Copies are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document room located at the Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Dated at Rockville, Maryland, this 28th day of January 1997.

For the Nuclear Regulatory Commission.

John B. Hickman,

Project Manager, Project Directorate III–3, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 97–3462 Filed 2–11–97; 8:45 am]

BILLING CODE 7590–01–P

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Thermal Hydraulic Phenomena; Notice of Meeting

The ACRS Subcommittee on Thermal Hydraulic Phenomena will hold a meeting on February 19, 1997, Room T–2B3, at 11545 Rockville Pike, Rockville, Maryland.

Portions of the meeting may be closed to public attendance pursuant to 5 U.S.C. 552b(c)(4), which authorizes closure of meetings to protect proprietary information, and 5 U.S.C. 552b(c)(9)(B), which authorizes closure of meetings to protect information the premature disclosure of which would be likely to significantly frustrate implementation of a proposed agency action.

The agenda for the subject meeting shall be as follows:

Wednesday, February 19, 1997—8:30 a.m. until the conclusion of business

The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions for deliberation by the full Committee, regarding technical issues associated with AP600 test data generated at the ROSA and Oregon State University APEX test facilities. The Subcommittee may hear separate presentations by representatives of the NRC staff and the Westinghouse Electric Corporation regarding the test data.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

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

Dated: February 6, 1997.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 97–3461 Filed 2–11–97; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 17, 1997, through January 31, 1997. The last biweekly notice was published on January 29, 1997 (62 FR 4341).

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

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

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

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

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

By March 14, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: November 26, 1996

Description of amendment request: The proposed amendment would change the definition of ''Primary Containment Integrity,'' Note 6 on Table 3.2.A, correct a typographical error on Table 3.2 D, correct Table 3.2.F to reflect modifications to the plant and changes to Bases sections 3/4.6G and 3/4.7.A. These changes are considered administrative and have no effect on plant design, safety limit settings or plant system operation.

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

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

The proposed administrative changes involving typographical errors, additions for clarity and consistency, and updating the Bases do not affect plant design, safety limit settings, or plant system operation and, therefore, do not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

The changes to instrument numbers and type do not change the parameters being surveyed or the number of operable channels for these parameters. These changes do not modify or add any initiating parameters and do not affect plant design, safety limit settings, or plant system operation. Therefore, these instrument changes do 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.

These proposed changes do not involve any potential initiating events that would create any new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

These changes do not affect any safety analysis assumptions, system operation, structures, potential initiating events or safety limits. Therefore, it is concluded that the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Patrick D. Milano, Acting

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

Date of amendment request: January 24, 1997

Description of amendment request: The proposed amendment will update the Safety Limit Minimum Critical Power Ratio (SLMCPR) in Technical Specification (TS) 2.1.2 and the associated Bases section to reflect the results of the latest cycle-specific calculation performed for the Pilgrim Nuclear Power Station Operating Cycle 12. In addition, the values provided in Note 5 of Table 3.2.C.1, which are based on the SLMCPR values, have been revised as a result of the changes to the SLMCPR value.

Basis for proposed no significant hazards 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: 11.The proposed technical specification changes do not involve a significant increase in the probability of an accident previously evaluated.

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

The basis of the MCPR [minimum critical power ratio] Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR preserves the existing margin to transition boiling, and the probability of fuel damage is not increased.

The basis of the MCPR criteria that define a limiting rod pattern is to ensure the SLMCPR is not violated in the event a control rod is fully withdrawn from the core. The new MCPR criteria that define a limiting rod pattern continue to ensure the SLMCPR is not violated in the event a control rod is fully withdrawn from the core. These new criteria do not change the method of operating the plant and have no effect on the probability of an accident initiating event or a transient.

Therefore, the proposed 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 result only from a revised method of analysis for the Cycle 12 core reload. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The new MCPR criteria that define a limiting rod pattern continue to ensure the SLMCPR is not violated in the event a control rod is fully withdrawn from the core.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Patrick D. Milano, Acting

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

Date of amendment request: January 29, 1997

Description of amendment request: The proposed change adds a new entry 3.0.5 to the plant Technical Specifications (TS) to provide specific guidance for returning equipment to service under administrative control for the sole purpose of performing testing to demonstrate operability.

Basis for proposed no significant hazards 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.

The proposed change does not affect the operation or design of the plant in any way. Operation of plant equipment under this change will not differ in any way from its normal operational mode. The normal operation of plant equipment is not a precursor to any accident. The purpose of tests performed using this change are to demonstrate that required automatic actions are carried out. Equipment will be operated under administrative control for only a short period of time. Personnel will be immediately available to take appropriate manual action if it should be required. Therefore operation of equipment under this change is not expected to increase the probability or consequences of an accident previously evaluated.

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

The proposed testing allowance does not involve any physical alterations or additions to plant equipment or alter the manner in which any safety-related system performs its function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3.

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

Equipment will be operated under administrative control for only a short period of time. Personnel will be immediately available to take appropriate manual action if it should be required. The purpose of the testing is to restore required equipment to an OPERABLE state which increases the automatic protection available and reduces the reliance on the compensatory measures provided by ACTION statements. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

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

NRC Project Director: Mark Reinhart, Acting

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: October 24, 1996

Description of amendment request: The proposed amendment to the Perry Nuclear Power Plant Technical Specifications revises those specifications associated with the Minimum Critical Power Ratio (MCPR) Reactor Core Safety Limit. The revision would increase the MCPR Safety Limit values to make them more conservative.

Basis for proposed no significant hazards determination: The NRC staff provides its analysis of the issue of no significant hazards consideration below:

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

There is no change to any plant equipment, and increasing the MCPR Safety Limit is more conservative. Therefore, the proposed change does not significantly increase 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.

There are no physical changes to the plant, and increasing the MCPR Safety Limit is more conservative. Therefore, the proposed changes do 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 MCPR Safety Limit values are more conservative, and were calculated using NRC approved methods. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The staff has reviewed the amendment request and the licensee's no significant hazards consideration determination. Based on the review and the above discussions, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

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

NRC Project Director: Gail H. Marcus

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 amendment request: August 19, 1996

Description of amendment request: The proposed amendments revise the steam generator (SG) repair criteria in the Byron, Unit 1, and Braidwood, Unit 1, Technical Specifications (TS). These revisions, if approved, would continue the use of the voltage-based SG tube repair criteria added by Amendment No. 77, dated November 9, 1995, to the Byron 1 TSs and by Amendment No. 69, dated November 9, 1995, to the Braidwood 1 TSs. The subject voltage-based repair criteria are applicable only for a specific form of SG tube degradation identified as outer diameter stress corrosion cracking (ODSCC), which is confined entirely within the thickness of the SG tube support plates (TSPs). Specifically, the pending amendments for both units would continue for one more operating cycle, the present use of a lower voltage repair limit of 3.0 volts on the hot leg side of the SGs using the Locked-Tube model. The cold leg side of the SGs and certain hot leg side tube/TSP intersections (e.g., dented SG tube intersections) would continue to be repaired using the Free-Span model. The proposed amendments are needed because the applicability of the revised voltage-based SG tube repair criteria for ODSCC which were added in the prior amendments cited above, was limited to only one full operating cycle for Braidwood 1 ending in spring 1997 and for the operating cycle ending in late 1997 for Byron 1.

Additionally, the inspection and reporting requirements added to the Byron 1 and Braidwood 1 TSs by the prior amendments cited above, would also be continued for one more operating cycle for both units. The maximum permissible value of the iodine-131 concentration in the primary coolant in both the Byron 1 and Braidwood 1 TSs remains unchanged at 0.35 microcuries per gram of coolant. Finally, the Bases sections in the Byron 1 and Braidwood 1 TSs are proposed to be revised to introduce the terminology associated with the Locked-Tube SG tube model and that of the Free-Span model.

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

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

This amendment request proposes to renew the SG tube plugging/repair criteria previously approved by the NRC in Amendments 69 and 77 to Braidwood and Byron Technical Specifications, respectively.

The previously evaluated applicable accidents are steam generator tube burst and Main Steam Line Break (MSLB). The postulated MSLB outside of containment but upstream of the Main Steam Isolation Valve (MSIV) represents the most limiting radiological condition relative to the IPC. The potential impact on public health and safety as a result of renewing the SG tube interim plugging criteria contained in the current Braidwood and Byron Technical Specifications is very low as discussed below. Tube burst due to predominantly axially oriented ODSCC at the TSP intersections is precluded during normal operating plant conditions since the tube support plates are adjacent to the degraded regions of the tube in the tube-to-tube support plate crevices.

During accident conditions, i.e., MSLB, the tubes and TSP may move relative to each other. This can expose the crack length portion to free-span conditions. Testing has shown that the burst pressure correlates to the crack length that is exposed to the free-span, regardless of the length that is still contained within the TSP bounds.

Therefore, a more appropriate methodology has been established for addressing leakage and burst considerations. This methodology is based on limiting potential TSP displacements (Locked-Tube Model Intersections) during postulated MSLB events, thus reducing the free-span exposed crack length to minimal levels. The tube expansion process employed in conjunction with this tube plugging criteria is designed to provide postulated TSP displacements that result in negligible tube burst probabilities due to the minimal free-span exposed crack lengths. The tube expansions were performed during the first outage that the 3.0 volt IPC was applied (Braidwood refuel outage A1R05 -Fall 1995, and Byron midcycle outage B1P02 - Fall 1995). These expansions will be inspected in accordance with an eddy current inspection probe that is sensitive to axial and circumferential indications. This program will ensure the integrity of the expansions for the additional cycle of operation. It has been demonstrated that axial indications in the expansion region will not result in a reduction of the load carrying capability of the expanded tubes.

Thermal hydraulic modeling was used to determine TSP loading during MSLB conditions. A safety factor was conservatively applied to these loads to envelope the collective uncertainties in the analyses. Various operating conditions were evaluated and the most limiting operating condition was used in the analyses. Additional models were used to verify the thermal hydraulic results.

Assessment of the tube burst probability for the Locked-Tube Model Intersections was based on a conservative assumption that all hot-leg TSP intersections (32,046) contained through wall cracks equal to the postulated TSP displacement and that the crack lengths were located within the boundaries of the TSP. Alternatively, it was assumed that all hot-leg TSP intersections contained through wall cracks with lengths equal to the thickness of the TSP. The postulated TSP motion was conservatively assumed to be uniform and equal to the maximum displacement calculated.

The total burst probability for all 32,046 through wall indications, given a uniform MSLB TSP displacement of 0.31 inches, was calculated to be $1x10^{-5}$. This is a factor of 1000 less than the GL 95-05 burst probability limit of $1x10^{-2}$. Therefore, the functional design criteria for tube expansion was to limit the TSP motion to 0.31'' or less. However, the design goal for tube expansion limits the TSP MSLB motion to less than 0.1''. This design goal results in a total tube burst probability of $1x10^{-10}$ for all 32,046 postulated through wall indications. Additional tubes were expanded to provide redundancy for the required expansions.

The structural limit for the Locked-Tube Model Intersection SG tube repair criteria was based on axial tensile loading requirements to preclude axial tensile severing of the tube. Axially oriented ODSCC does not significantly impact the axial tensile loading of the tube. Based on the current voltage distributions and growth rates, Monte Carlo projections were performed for Braidwood Unit 1 and Byron Unit 1 for the additional cycle of operation that this proposed amendment is requesting. The End of Cycle (EOC) voltage projections for Braidwood Unit 1 Cycle 7 predict that the maximum voltage to be seen will be less than 10.5 volts. The number of indications predicted greater than ten volts at the end of Cycle 7 for Braidwood Unit 1 is 0.3. The EOC voltage projections for Byron Unit 1 Cycle 9 predict that the maximum voltage to be seen will be less than 13.5 volts. The number of indications predicted greater than ten volts at the end of Cycle 9 for Byron Unit 1 is 4.59.

Using a tensile rupture probability for a ten volt indication of $3x10^{-6}$, the probability of tensile rupture from the predicted 0.3 indications at Braidwood is $1\text{-}(1\text{-}3x10^{-6})^{0.3} = 9.0x10^{-7}$. The probability of tensile rupture from the predicted 4.59 indications at Byron is $1\text{-}(1\text{-}3x10^{-6})^{4.59} = 1.38x10^{-5}$. Both of these probabilities result in a negligible contribution to the total burst probability when compared to the $1x10^{-2}$ GL 95-05 limit.

Cellular corrosion is a more limiting mode of degradation at the TSPs with respect to affecting the tube structural limit. Tensile tests that measure the force required to sever a tube with cellular corrosion and uncorroded cross sectional areas are used to establish the lower bound structural limit. Based upon these tests, a lower bound 95% confidence level structural voltage limit of 37 volts was established for cellular corrosion. This limit meets the Regulatory Guide (RG) 1.121, ''Basis for Plugging Steam Generator Tubes,'' structural requirements based upon the normal operating pressure differential with a safety factor of 3.0 applied. Due to the limited database supporting this value, the structural limit was conservatively reduced to 20 volts. Accounting for voltage growth and Non-Destructive Examination (NDE) uncertainty, the full IPC upper limit exceeds ten volts. However, for added conservatism a single voltage repair limit of 3.0 volts for the Locked-Tube Model Intersection indications is specified in the current plugging/repair criteria. All indications at the Locked Tube Model Intersections with bobbin coil probe voltages greater than 3.0 volts will be plugged or repaired.

The free-span tube burst probability must be calculated for the indications at the Free-Span Model Intersections. The total burst probability must be within the requirements of GL 95-05. The free-span structural voltage limit is calculated using correlations from the database described in GL 95-05, with the inclusion of the recent Byron, Braidwood, and South Texas tube pull results. The structural limit for the Free-Span Model Intersections is 4.745 volts. The lower voltage repair limit for the indications at the Free-Span Model Intersections continues to be 1.0 volt. The upper voltage repair limit for the indications at the Free-Span Model Intersections will be calculated in accordance with GL 95-05.

Since IPC will not be applied to indications at the Flow Distribution Baffle (FDB), no leakage or burst analyses are required for these indications.

Per GL 95-05, MSLB leak rate and tube burst probability analyses are required to be performed prior to returning the unit to power. The results of these analyses are to be included in a report to the NRC within 90 days of restart. If allowable limits on leak rates and burst probability are exceeded, the results are to be reported to the NRC and a safety assessment of the significance of the results is to be performed prior to returning the SGs to service.

A site specific calculation has determined the site allowable leakage limit for Braidwood and Byron. These limits use the recommended Dose Equivalent Iodine-131 transient spiking values consistent with NUREG-0800, ''Standard Review Plan'' and ensure site boundary doses are within a small fraction of the 10 CFR 100 requirements.

The projected leakage rate calculation methodology described in WCAP-14046, ''Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria,'' and WCAP-14277, ''SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections,'' will be used to calculate the EOC leakage. This method includes a Probability of Detection (POD) value of 0.6 for all voltage amplitude ranges and uses the accepted leak rate versus bobbin voltage correlation methodology (full Monte Carlo) for calculating the leak rate, as described in GL 95-05. The database used for the leak and burst correlations is consistent with that described in GL 95-05 with the inclusion of the Byron Unit 1, Braidwood Unit 1, and South Texas tube pull results. The EOC voltage distribution is developed from the POD adjusted beginning-of-cycle (BOC) voltage distributions and uses Monte Carlo techniques to account for variances in growth and NDE uncertainty.

The Electric Power Research Institute (EPRI) leak rate correlation has been used. This correlation is based on free-span indications that have burst pressures above the MSLB pressure differential. There is a low but finite probability that indications may burst at a pressure less than MSLB pressure. With limited TSP motion for the Locked-Tube Model Intersections, the tube is constrained by the TSP and tube burst is precluded. However, the flanks of the crack may open up to contact the Inside Diameter (ID) of the TSP hole and result in a primary-to-secondary leak rate potentially exceeding that obtained from the EPRI correlation. This phenomenon is known as an Indication Restricted from Burst (IRB) condition.

ComEd has performed laboratory testing to determine the bounding leak rate obtainable in an IRB condition (6.0 gallons per minute). The bounding leak rate value was then applied to a leak rate calculation methodology that accounts for the MSLB leak rate contribution from IRB indications to the total leak rate calculated as described above. Results indicate that the IRB contribution to the total leak rate value is negligible. However, ComEd will conservatively add a leakage contribution due to IRBs in addition to the leakage calculated in accordance with GL 95-05. When this is done, the dose at the site boundary resulting from the predicted leakage will be a small fraction (less than 10%) of the 10 CFR 100 limits.

Modification of the Braidwood and Byron TS to clarify application of the proposed tube plugging/repair criteria is purely administrative and will not have any effect on the probability or consequences of an accident previously evaluated.

Operating experience over the last cycle with this plugging criteria applied has not revealed any unpredicted or unusual effects.

For these reasons, renewal of the current Braidwood and Byron tube plugging criteria does not adversely affect SG tube integrity and results in acceptable dose consequences. By effectively eliminating tube burst at the Locked-Tube Model TSP intersections, the likelihood of a tube rupture is substantially reduced and the probability of occurrence of an accident previously evaluated is reduced.

This conclusion is not affected by foreign or domestic plant SG experiences (NRC Information Notice 96-09 and its supplement). As the following evaluation shows, these experiences are not relevant to Braidwood or Byron.

A foreign unit detected eddy current signal distortions in one area of the top TSP during a 1995 inspection. The steam generators had been chemically cleaned in 1992. Visual inspection showed that a small section of the top TSP had broken free and was resting next to the steam generator tube bundle wrapper. The support plate showed indications of metal loss.

The chemical cleaning process used by the foreign unit was developed by the utility and differs significantly from the modified EPRI/SGOG process performed at Byron Unit 1 in 1994. The foreign chemical cleaning process, coupled with the specific application of the process, resulted in TSP corrosion of up to 250 mils compared to a maximum of 2.16 mils (11 mils maximum allowed) measured at Byron. During the Byron eddy current inspection performed after the chemical cleaning, no distortion of the tube support plate signals was reported. Therefore, these differences in cleaning processes imply that this foreign experience is irrelevant to the effects of the chemical cleaning process on the TSPs at Byron. Chemical cleaning of the SGs has not occurred at Braidwood.

A number of units have experienced TSP cracking associated with severe tube denting due to TSP corrosion at the tube-to-TSP crevice. WCAP-14273, Section 12.4, shows that a diametral reduction of a SG tube of 0.065 inches is required to develop stress levels above yield in the TSP ligaments at dented intersections. The bobbin voltage range associated with a one mil radial dent is twenty to twenty-five volts.

Although Braidwood Unit 1 and Byron Unit 1 have not seen corrosion induced denting, a 0.610 inch diameter bobbin coil probe will be used as a go/no-go gauge to assess dents at the Locked-Tube Model Intersections, if they occur in the future. If a tube has a dent at a Locked-Tube Model TSP intersection that fails to pass the go/no-go test probe, IPC will not be applied to that intersection. In addition, if the dent is determined to be corrosion induced, the Free-Span Model repair criteria will be applied to the intersections adjacent to the dented intersection. IPC repair limits will not be applied to tubes with dents greater than 5.0 volts since dent signals of this magnitude could mask a 1.0 volt ODSCC signal. Tube intersections with corrosion induced dents greater than 5.0 volts and the intersections adjacent to such an intersection were not selected for tube expansion to preclude adverse effects of the failure of such a tube on limiting TSP displacement. If corrosion induced denting, either greater than 5.0 volts or such that the tube is unable to pass a 0.610 inch diameter bobbin coil probe, are detected at an intersection adjacent to an expanded intersection, the dented intersection will be inspected by an EPRI developed technique to determine if the TSP is cracked. If a crack-like indication is identified in a TSP, a plus point inspection will be conducted per the EPRI TSP program. If the plus point inspection verifies the existence of a crack-like indication, the effect of that indication on TSP displacement will be evaluated. If this evaluation shows that TSP displacement would be greater than 0.1 inches during a MSLB event, the effected area will either be mechanically corrected or the Free-Span Model criteria will be applied to the affected area. Based on the information presented above, the SG tube denting experience at other plants is not relevant to Braidwood or Byron.

A foreign utility's SGs have experienced cracking at the top TSP. The cause of the cracking appears to be the configuration of the single anti-rotation device, connected between the SG shell and wrapper, and the wrapper internals. The single anti-rotation device carries the full load associated with the wrapper to shell motion. This rotational load is believed to be transferred to the TSP via the wrapper internals. The Byron/Braidwood Unit 1 SG design (D-4) uses three anti-rotation devices to spread the rotational load. The D-4 wrapper internals are configured such that this load is not directly transmitted to the TSP.

No top TSP cracking has been detected at Braidwood Unit 1 or Byron Unit 1 and very few (<1%) of the ODSCC indications in the SG tubes at Braidwood and Byron, to date, have been at the top TSP elevation. Nevertheless, an analysis was performed to assess the impact of cracking of the top TSP. The results show an increase in the deflection of the top TSP for a very limited number of tubes to greater than the 0.10'' limit used in the 3.0 volt IPC analysis. The deflections of the lower support plates also increased, but remain within the 0.10'' limit. Thus, a large majority of the Locked-Tube Model indications continue to be bounded by the existing analysis even with a cracked top TSP. The Locked-Tube Model repair criteria will not be applied to any SG tube ODSCC indication where the TSP has been shown to be displaced by more than 0.1 inches during accident conditions.

In response to these experiences at foreign and domestic utilities, ComEd developed an inspection plan for the SG internals to identify if indications detrimental to the load path components existed. This inspection plan was carried out at Braidwood during refueling outage A1R05 (Fall 1995) and at Byron during the midcycle outage B1P02 (Fall 1995) and refuel outage B1R07 (Spring 1996). These inspections revealed no degradation of the SG load path components necessary to support implementation of the 3.0 volt IPC. Inspections will be performed during the upcoming refuel outages at Braidwood Unit 1 and Byron Unit 1 to further ensure the integrity of the SG load path components necessary to support implementation of the 3.0 volt IPC.

A domestic utility reported several distorted TSP signals over the past three refueling outages' SG tube inspections. It was determined that these signals were associated with the TSP geometry in an area where an access cover is welded to the TSP. These signal distortions are not attributed to TSP cracking or degradation. Since the distorted signals were due to TSP geometry which did not indicate or result in a defect of the TSP, there is no increase in the probability or consequences of an accident previously evaluated due to Braidwood Unit 1 and Byron Unit 1 steam generator TSP geometries which may result in distorted eddy current signals.

One foreign unit observed a dislocation of the tube bundle wrapper when they were unable to pass sludge lancing equipment through a hand hole in the wrapper. The dislocation appears to be a result of improper attachment of the wrapper to the support structure. SG sludge lance operations have been successfully performed at Braidwood Unit 1 and Byron Unit 1 which indicates that no problem with the wrapper attachment exists. The foreign unit's wrapper support design is significantly different than that used on Braidwood Unit 1 and Byron Unit 1. Therefore, a similar wrapper dislocation will not occur and the foreign experience is not applicable to Braidwood or Byron. An inspection was conducted during the last Braidwood Unit 1 and Byron Unit 1 refueling outages which verified this conclusion.

ComEd will continue to apply a maximum primary-to-secondary leakage limit of 150 gallons per day (gpd) through any one SG at Braidwood and Byron to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing operational leakage limits that require plant shutdown are based on detecting a free-span crack prior to it resulting in primary-to-secondary operational leakage which could potentially develop into a tube rupture during faulted plant conditions. The 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected single crack leak associated with the longest permissible free-span crack length.

Therefore, the proposed amendment does not result in any significant increase in the probability or consequences of an accident previously evaluated within the Braidwood and Byron Updated Final Safety Analysis Report (UFSAR).

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

This amendment request proposes to renew the SG tube plugging/repair criteria previously approved by the NRC in Amendments 69 and 77 to Braidwood and Byron Technical Specifications, respectively.

Renewal of the proposed steam generator tube plugging criteria with tube expansion does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations as ODSCC does not extend beyond the thickness of the tube support plates and IPC is not allowed to be applied to indications that extend beyond the thickness of the tube support plate. Neither a single nor multiple tube rupture event would be expected in a SG in which the plugging criteria has been applied.

The tube burst assessment involves a Monte Carlo simulation of the site specific voltage distribution to generate a total burst probability that includes the summation of the probabilities of one tube bursting, two tubes bursting, etc. For the Locked-Tube Model TSP Intersections, the maximum total probability of burst, by design, is estimated to be $1x10^{-10}$ with all tube expansions functional. The burst probability for the Free-Span Model TSP intersections will be dependent on the number and size of indications at these applicable intersections. The total burst probability will be within the limit specified in GL 95-05.

Accounting for the unlikely event of a failure of the expanded tubes, a sufficient number of redundant expansions exist to ensure that the burst probability remains below $1x10^{-5}$. This includes the conservative assumption that all 32,046 hot-leg TSP intersections contain through wall indications. This level of burst probability is considered to be negligible when compared to the GL 95-05 limit of $1x10^{-2}$.

In addressing the combined effects of a Loss Of Coolant Accident (LOCA) during a Safe Shutdown Earthquake (SSE) on the SG as required by General Design Criteria (GDC) 2, it has been determined that tube collapse may occur in the steam generators at some plants. The tube support plates may become deformed as a result of lateral loads at the wedge supports located at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. The resulting pressure differential on the deformed tubes may cause some of the tubes to collapse. There are two issues associated with SG tube collapse. First, the collapse of SG tubing reduces the Reactor Coolant System (RCS) flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase the Peak Clad Temperature (PCT). Second, there is a potential that partial through wall cracks in the SG tubes could progress to through wall cracks during tube deformation or collapse. The tubes subject to collapse have been identified via a plant specific analysis and are excluded from application of any voltage-based criteria. This analysis is included in revision 3 to WCAP-14046 which was submitted to the NRC June 19, 1995.

Modification of the Braidwood and Byron Technical Specifications to clarify application of the proposed tube plugging/repair criteria is purely administrative and will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Operating experience over the last cycle with this plugging criteria applied has not revealed any unpredicted or unusual effects.

SG tube integrity will continue to be maintained following renewal of the 3.0 volt IPC voltage repair limit through inservice inspection, tube repair and primary-to-secondary leakage monitoring. By effectively eliminating tube burst at the Locked-Tube Model TSP Intersections, the potential for multiple tube ruptures is essentially eliminated.

ComEd has evaluated industry experiences with TSP degradation, eddy current signal distortions, and component misalignment. Eddy current signal distortions due to TSP geometry are not indicative of TSP degradation and do not result in any kind of new or different accident.

The component misalignment experienced by one unit is not applicable to Braidwood Unit 1 or Byron Unit 1 and, thus, will not result in any kind of new or different accident. Specific limitations, as discussed in response to Question 1, will be applied to indications at the Locked-Tube Model Intersections which contain dents. These limitations ensure that the integrity of the SG tubes is maintained consistent with the current analyses should tube denting or TSP cracking occur.

Therefore, renewal of the current tube plugging/repair criteria at Braidwood Unit 1 and Byron Unit 1 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 use of the voltage-based, bobbin coil, tube support plate plugging criteria with tube expansion at Braidwood Unit 1 and Byron Unit 1 is demonstrated to maintain SG tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture.

Reducing the probability or the consequences of steam generator tube rupture is accomplished by determining an eddy current inspection voltage value which represents a limit for leaving an axial, crack-like indication at an in service SG tube TSP intersection. Tubes with ODSCC voltage indications beyond this limiting value must be removed from service by plugging or repaired by sleeving. Implementation of a 3.0 volt IPC voltage repair limit for the Locked-Tube Model Intersections has been evaluated and shown not to present a credible potential for a steam generator tube rupture event during normal or faulted plant conditions, even with worst case assumptions. The total tube burst probability will include a contribution from the indications at the Locked-Tube Model Intersections and from indications at the Free-Span Model Intersections. The projected EOC voltage distribution of crack-like indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions such that radiological consequences are not adversely impacted.

Addressing RG 1.83 considerations, implementation of the increased Locked-Tube Model Intersection bobbin coil voltage-based repair criteria is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization and a 100% eddy current inspection sample size at the affected TSP elevations.

For the leak and burst assessments, the population of indications in the EOC voltage distribution is dependent on the POD function. The purpose of the POD function is to account for new indications that may develop over the cycle, and to account for indications not identified by the data analyst. In implementing this proposed IPC renewal, ComEd will continue to use the conservative GL 95-05 POD value of 0.6 for all voltage amplitude ranges.

Modification of the Braidwood and Byron Technical Specifications to clarify application of the proposed tube plugging/repair criteria is purely administrative and will not reduce any safety margins.

Operating experience over the last cycle with this plugging criteria applied has not revealed any unpredicted or unusual effects.

Implementation of the TSP elevation repair limits will decrease the number of tubes which must be repaired. Installation of steam generator tube plugs or sleeves reduces the RCS flow margin. Thus, implementation of the IPC will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging or sleeving.

As discussed previously, ComEd has evaluated industry experiences with TSP degradation, eddy current signal distortions, and component misalignment. Eddy current signal distortions at tube support plates will be evaluated to attempt to determine the cause of the distortion. A signal distortion alone will not result in reduction in the margin of safety. The foreign unit that experienced the component misalignment was of a significantly different design than the Braidwood Unit 1 and Byron Unit 1 steam generators. Analysis of the design differences shows that component misalignment of that type is not applicable to Braidwood Unit 1 or Byron Unit 1 and, thus, will not result in a reduction in the margin of safety. An inspection was conducted during the last Braidwood Unit 1 and Byron Unit 1 refueling outages which verified this conclusion.

Specific limitations, as discussed previously, will be applied to indications at the Locked-Tube Model Intersections which contain dents. These limitations conservatively treat indications as free-span to ensure that the integrity of the SG tubes is maintained consistent with current analyses should tube denting or TSP cracking occur. Application of the 3.0 volt Locked-Tube Model Intersection IPC and the 1.0 volt Free-Span Model Intersection IPC at Braidwood Unit 1 and Byron Unit 1, with the limitations specified, will not result in a reduction in a margin of safety.

Thus, the implementation of this amendment does not result in 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 requested amendments involve no significant hazards consideration.

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.

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

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: January 6, 1997

Description of amendment request: The proposed amendment would clarify and maintain consistency between the operability requirements for protective instrumentation and associated automatic bypass features.

Basis for proposed no significant hazards 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 of the following:

The proposed changes are administrative in nature and do not affect the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and represents sufficient requirements for the continued assurance and reliability of the RPS and Rod Block Instrumentation equipment, which is assumed to operate in the safety analysis, or provides continued assurance that specified parameters associated with RPS and Rod Block Instrumentation remain within their acceptance limits. Therefore, these changes will not affect the probability or consequences of a previously evaluated accident.

The RPS and Rod Block Instrumentation related to this proposed amendment is not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the probability of any accident previously evaluated is not affected by the proposed amendment.

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

The proposed changes are administrative in nature and serve to maintain consistent and clear requirements for operability as specified in the Technical Specifications for the Limiting Conditions for Operation and Surveillance Requirements for the RPS and Rod Block Instrumentation. No new modes of operation or changes to any plant equipment are proposed by the proposed amendment request. The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities. The proposed changes maintain the present level of operability; and therefore, the proposed changes do not create the possibility of a new or different kind of accident than any previously evaluated.

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

The proposed changes are administrative in nature and do not affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes have been evaluated and found to be acceptable for use at Dresden and at Quad Cities based on RPS and Rod Block Instrumentation system design, safety analysis requirements and operational performance. Since the proposed changes are administrative in nature and maintain necessary levels of the RPS and Rod Block reliability, the proposed changes do not involve a reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of the RPS and Rod Block Instrumentation System which is required to mitigate accident conditions; therefore, the proposed changes do not involve a reduction in the margin of safety.

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

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: October 22, 1996

Description of amendment request: The proposed amendments would allow continued plant operation at elevated Containment Lower Compartment temperatures between 125° and 135° F for a period not to exceed 72 cumulative hours.

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

1. Does the proposed amendment involve a significant increase in the probability or consequences of any accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report]?

The increase in maximum Containment Lower Compartment temperature will not change the operation of any equipment which is important to safety. All components and instruments will continue to perform as designed in the higher temperature environment for the period that the revised Technical Specification allows. This temperature increase will not impact the ability of any component or instrument to perform its function in the event of an accident. Therefore, the probability of an accident is not impacted. The increased temperature will cause a decrease in the air mass in lower containment. This change has been evaluated for impact on containment temperature and pressure in accident conditions. The air mass change is conservative for peak containment pressure since the air mass is decreased. Maximum containment temperatures during a postulated accident are slightly increased as a result of higher initial Containment Lower Compartment temperature. The increase in peak temperature remains within the allowable values and thus does not increase the probability or consequence of an accident. The minimum containment pressure as a result of steam condensation in containment is lowered as a result of the decreased air mass in containment. Due to the conservative assumptions made in modeling containment for minimum pressure response, this change has no impact on the accident analysis.

Based on the analysis of the bounding accidents that may be impacted by increased Containment Lower Compartment temperature and the review of the effect of the increased temperature on components in lower containment, it is determined that the probability and consequence of any analyzed accident is unchanged as a result of this change.

2. Does the proposed amendment create the possibility of a new or different kind of accident not previously evaluated?

The revised maximum Containment Lower Compartment temperature will not change any systems or operations procedures except to procedurally respond should Containment Lower Compartment temperature remain elevated for a period near the revised limiting period. The response of the systems and components are unaffected by this change. All instruments are qualified for the revised service conditions and will perform in the same manner as before. Normal operation and transient response will remain unchanged. Review of previously analyzed accidents show that no new transients are created as a result of this change. Based on this review there are no new or different accidents made possible by this change.

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

The amendment could potentially affect the containment system. The operation and analysis of the reactor coolant system and fuel are unaffected by this change. The maximum containment temperature is slightly increased while the maximum containment pressure is decreased. The minimum containment pressure could be slightly decreased and minimum containment temperature is unaffected. All these parameters have been reviewed and determined to be within assumptions made in these analyses. The accident transient analyses are unaffected beyond these small changes and remains acceptable in all cases. Therefore, the margin of safety is unaffected by this amendment.

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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina 28223-0001

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: January 6, 1997

Description of amendment request: The proposed amendments would allow a one-time revision to Technical Specifications 3.6.1.1, 3.6.1.2, 3.6.1.8, and 3.6.1.9 to allow operation of the Containment Purge Ventilation System (VP) during Modes 3 and 4 following the steam generator (SG) replacement outage. This one-time revision would be necessary due to respiratory hazardous gases released during heatup after the replacement of the SGs. The VP system would be used to remove the hazardous gases.

Basis for proposed no significant hazards 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 activity does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The VP [Containment Purge Ventilation] System has no interfaces with any primary system, secondary system, or power transmission system. It has no interfaces with any reservoir of radioactive gases or liquids. None of the systems listed above are modified by the activity. In summary, no ''accident initiator'' is affected with the proposed operation of the VP System in Modes 3 and 4. For this reason, the activity does not involve an increase in the probability of an accident previously evaluated.

Analyses have been performed to determine upper bounds to the source term, the offsite doses, and the Control Room dose. The results of that analyses are reported above. Both the source term and the doses were found to be significantly lower than the results of the corresponding design basis analyses. In addition, it has been determined that with no credit taken for any heat transfer from the fuel and cladding to the moderator channels, that sufficient time would exist for the operators to initiate recovery of flow from the ECCS [Emergency Core Cooling System] to the reactor core. The flow required from the ECCS to maintain the core in a coolable geometry was found to be well within the capacity of any one ECCS pump. Furthermore, it was determined that convective heat transfer to steam would be sufficient to prevent release of significant source term or a significant degree of fuel damage.

For the above reasons, it is determined that operation of the VP System in Mode 3 or 4 immediately following the steam generator replacement outage does not involve a significant increase in either the probability or the consequences of an accident previously evaluated.

2. The activity does not create the possibility of a new or different type of

accident from any accident previously evaluated.

As discussed above, no ''accident initiators'' are affected by the proposed activity. Operation of the VP System proposed for Modes 3 and 4 will be the same as that routinely carried in other modes of operation. For these reasons, the activity will not create the possibility of a new or different type of accident from any previously evaluated.

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

Margin of safety is associated with confidence in the ability of the fission product barriers (the fuel and fuel cladding, the Reactor Coolant System pressure boundary, and the containment) to limit the level of radiation doses to the public. The proposed operation of the VP System will occur at the end of an extended outage. The level of decay heat and activity in the reactor is very low compared to the level of decay heat and activity associated with full power operations. For this reason, the likelihood of damage to the fuel following a DBLOCA [design basis loss-of-coolant accident] occurring during the proposed purging is reduced, as determined above. Both offsite doses and doses to the Control Room were found to be small compared to the limits of 10 CFR [Part] 100 and GDC [General Design Criterion] 19. For these reasons, the activity 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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina 28223-0001

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: January 13, 1997

Description of amendment request: The proposed amendments would implement the performance-based containment leak rate testing requirements of 10 CFR Part 50, Appendix J, Option B, for Type A testing.

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components that penetrate the primary containment, do not exceed the allowable leakage rate values specified in the Technical Specifications and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety that impose a significant regulatory burden. Appendix J to 10 CFR Part 50, ''Primary Containment Leakage Testing for Water-Cooled Power Reactors,'' was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, ''Performance-Based Leak-Test Program.''

Based on the results of this study, the staff developed a performance based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B ''Performance-Based Requirements'' to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' was developed as a method acceptable to the staff for implementing Option B. Accordingly, the licensee has submitted, in its application dated January 13, 1997, proposed changes to the TS to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program.''

Basis for proposed no significant hazards 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 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Containment leak rate testing is not an initiator of any accident; the proposed change does not affect reactor operations or accident analysis, and has no significant radiological consequences. ... Therefore, this proposed change will not involve an increase in the probability or consequences of any previously-evaluated accident.

2. The proposed change will not create the possibility of any new accident not previously evaluated.

The proposed change does not affect normal plant operations or configuration, nor does it affect leak rate test methods. The test history at McGuire (two consecutive successful tests) provides continued assurance of the leak tightness of the containment structure.

3. There is no significant reduction in a margin of safety.

The proposed changes are based on NRC-accepted provisions, and maintain necessary levels of reliability of containment integrity. The performance-based approach to leakage rate testing recognizes that historically good results of containment testing provide appropriate assurance of future containment integrity; this supports the conclusion that the impact on the health and safety of the public as a result of extended test intervals is negligible. In addition, local leak[]rate testing will continue to provide assurances of overall containment integrity.

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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina 28223-0001

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: October 16, 1996

Description of amendment request: The amendment requests to change the Waterford 3 Technical Specifications Table 4.3-1 to expand the applicability for Core Protection Calculator operability and to allow for the application of a Cycle Independent Shape Annealing Matrix.

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

The proposed change will reduce the amount of non-conservatism presently allowed for linear power level, the CPC delta T power, and CPC nuclear power signals.

Changing the tolerance range from plus or minus 2% to between -0.5% and 10% between 15% and 80% RATED THERMAL POWER, except during physics testing, will allow more conservative settings than currently allowed. The consequences of an accident will be reduced due to the proposed change because it is less likely to be non-conservative in power.

This proposed change will allow use of Cycle Independent Shape Annealing Matrix (CISAM) elements. These elements will be validated, during startup testing, by monitoring the same parameters used for cycle specific shape annealing matrix (SAM) elements. If the CISAM is determined to be no longer valid, a cycle specific SAM will be calculated and used in the CPC's. In addition, use of CISAM gives better agreement throughout the cycle.

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

The proposed change to TS power calibration tolerance limits is conservative relative to the current TS requirement. CPC's cannot cause an accident and this change will not create the possibility of a new or different type accident. The changes ensures that the reactor will trip prior to the current condition due to higher CPC power.

As stated previously, CISAM modeling removes some of the uncertainty associated with axial shape and provides increased assurance that the CPC is appropriately modeling the core.

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

The proposed change to the TS reduces the amount of non-conservatism in safety system power indications and maintains the margin of safety for design basis events which take credit for the linear power level, the CPC delta T power, and CPC nuclear power signals.

CISAM will be validated each cycle during startup testing and must meet the same parameters as cycle specific SAM elements. Since CISAM has a better accuracy than the cycle dependent SAM, the margin of safety is improved.

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

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: November 27, 1996

Description of amendment request: The proposed change request would change the acceptance criteria for the individual cell voltage from 2.0v to 2.09v, change the surveillance frequency for battery specific gravities to implement the recommendations of IEEE 450-1995, delete surveillance requirement 4.7.B.4.d, add a clarifying phrase ''while on a float charge....'' where appropriate, and update the Basis to reflect these changes.

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

This request has been determined to involve No Significant Hazards in that it does not:

1. Involve a significant increase in the probability or consequences of an accident previous[ly] evaluated; (or)

The proposed change in ICVs [individual cell voltages] does not increase the probability of an accident previously evaluated, as it increases the required voltage for each ICV.

The proposed change in frequency does not increase the probability or consequences of an accident previously evaluated, as the change in the frequency of specific gravity testing is the result of industry experience gained over the years. The weekly reading of pilot cell specific gravity and cell voltage, along with the quarterly reading of all ICVs and a 10% sample of specific gravities from designated cells provides an acceptable means of determining cell operability as specified in IEEE 450-1995.

The proposed deletion of Technical Specification Surveillance Requirement 4.7.B.4.d only removes an unnecessary Technical Specification surveillance and is consistent with NUREG-1433, Standard Technical Specifications General Electric Plants, BWR/4, Revision 1, April 1995. No change to plant systems, components or operating conditions are associated with this change. Existing Technical Specification station and diesel generator battery inspection and testing requirements adequately verify battery operability and condition.

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

The proposed change does not create the possibility of a new or different kind of accident than previously evaluated, as the change only involves raising a required voltage, performing an existing surveillance on a different frequency, and removing an unnecessary annunciator surveillance requirement. The station battery and diesel generator battery low voltage annunciator setpoints do not meet any of the criteria codified in 10 CFR 50.36 for determining content of Technical Specifications and removal of surveillance requirement is consistent with NUREG-1433, Standard Technical Specifications General Electric Plants, BWR/4, Revision 1, April 1995. There is no change to hardware or operating conditions.

3. Involve a significant decrease in the margin of safety.

The proposed change to the ICV does not decrease the margin of safety, as increasing the required voltage actually increases the margin of safety. The proposed change to the frequency does not decrease the margin of safety as it continues to require testing and evaluation of the requisite surveillance points and implements requirements which have been determined to provide an adequate level of safety by the IEEE. The removal of Technical Specification surveillance requirements for the battery low voltage annunciator setpoints does not affect any plant systems, components or operating conditions.

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: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

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

NRC Project Director: Patrick D. Milano, Acting

Northern States Power Company, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: January 23, 1997, as revised by letter dated January 28, 1997.

Description of amendment request: The proposed amendment would make changes to Section 3.5/4.5.C of the technical specification (TS) bases to clarify the minimum residual heat removal (RHR) and residual heat removal service water (RHRSW) pump requirements for post-accident containment heat removal. In conjunction with the proposed amendment, the licensee requested NRC staff review and approval of an update to the design basis accident containment temperature and pressure response for the limiting single failure (loss of diesel generator) which results in minimum RHR and RHRSW pump availability.

Basis for proposed no significant hazards 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.

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

The proposed amendment will change TS bases to clarify the minimum RHR and RHRSW pump requirements for post-accident containment heat removal. The proposed amendment will also correct an error in a previous analysis on containment temperature and pressure response following a design basis accident (DBA) that was submitted for the NRC staff review on May 1, 1986. The proposed amendment does not affect the physical configuration of the plant or how it is operated. The licensee's analysis, using a new decay heat model, determined that the calculated maximum suppression pool temperature will be 2 degrees Fahrenheit greater (184 degrees Fahrenheit vs. 182 degrees Fahrenheit) than that predicted in its previous analysis, based on an earlier decay heat model, that was submitted for the NRC staff review on May 1, 1986. The licensee evaluated the effects of this increase on emergency core cooling system (ECCS) pump net positive suction head, wetwell attached piping, and environmental conditions in the ECCS pump rooms, and concluded that the change is acceptable. The consequences or probability of a previously evaluated accident will, therefore, not be significantly increased.

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

The proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendment does not affect the physical configuration of the plant or how it is operated. The proposed amendment revises the TS bases to clarify the minimum RHR and RHRSW pump requirements for post-accident containment heat removal.

(3) The proposed changes do not result in a significant reduction inthe margin of safety.

The proposed amendment will change TS bases to clarify the minimum RHR and RHRSW pump requirements for post-accident containment heat removal. The proposed amendment will also correct an error in a previous analysis on containment temperature and pressure response following a design basis accident (DBA) that was submitted for the NRC staff review on May 1, 1986. The proposed amendment does not affect the physical configuration of the plant or how it is operated. The licensee's analysis, using a new decay heat model, determined that the calculated maximum suppression pool temperature will be 2 degrees Fahrenheit greater (184 degrees Fahrenheit vs. 182 degrees Fahrenheit) than that predicted in its previous analysis, based on an ealier decay heat model, that was submitted for the NRC staff review on May 1, 1986. The licensee evaluated the effects of this increase on emergency core cooling system (ECCS) pump net positive suction head, wetwell attached piping, and environmental conditions in the ECCS pump rooms, and concluded that the change is acceptable. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment requests: December 9, 1996

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to revise the surveillance frequencies from at least once every 18 months to at least once per refueling interval (nominally 24 months) for the reactor trip system (RTS) and engineering safety features actuation systems (ESFAS) instrumentation channels, and make certain changes in trip setpoints and allowance values due to a setpoint methodology change in support of the calibration extensions. Channel operational tests (COTs) and trip actuating device operational tests (TADOTs) associated with these channels are also being extended. Revisions to the appropriate TS Bases are being revised to support the TS revisions.

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

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

The proposed TS channel calibration, COT, and TADOT interval increases from 18 to 24 months, the setpoint change, and the allowable value changes do not alter the intent or method by which the channel calibrations are conducted, do not alter the way any structure, system, or component functions, and do not change the manner in which the plant is operated. The calibration and maintenance histories indicate that the equipment will continue to perform satisfactorily with longer surveillance intervals. With the exception of the pressurizer water level - high instrument, no recurring surveillance or maintenance problems were identified for the RTS or ESFAS instrumentation channels.

The pressurizer water level instruments do not have a safety limit and are not credited in the DCPP safety analysis. The recurring surveillance problems were mainly due to calibration zero shift which is reflected in the statistically determined drift and in the proposed pressurizer water level high setpoint. The zero shift problem of these transmitters was a recurring problem with the calibration procedure. The procedures for calibrating these instruments have been revised to improve the repeatability of the surveillance activity.

The trip setpoint and allowable value changes for pressurizer water level - high are each in the more restrictive direction. The revised setpoint would tend to trip the reactor sooner than the present settings. These changes ensure that sufficient margin is maintained for the pressurizer water level to accommodate the channel statistical uncertainty resulting from a 30-month operating cycle.

A statistical analysis of channel uncertainty for a bounding 30-month operating cycle has been performed. There is sufficient margin between the existing TS limits and the licensing basis safety analysis limits to accommodate the channel statistical uncertainty resulting from a 30-month operating cycle. The existing margin between the TS limits and the safety analysis limits provides assurance that plant protective actions will occur as required. However, a change to the safety analysis limit is proposed in order to provide additional margin for the RCS loss of [f]low-low setpoint.

Westinghouse has evaluated the safety analysis limit for the RCS loss of flow-low setpoint and has determined that the limit can be changed from 87 percent of MMF to 85 percent of MMF with no impact on the probability and insignificant impact on the consequences of accidents already analyzed. The existing conclusions of the DCPP FSAR Update remain valid with the safety analysis limit change. Using the new safety analysis limit, sufficient margin exists between the TS limit and the safety analysis limit to accommodate the channel statistical uncertainty resulting from a 30-month operating cycle.

The proposed changes to the allowable values ensure that drift assumptions regarding the protection racks and direct input functions are met.

There are no known mechanisms that would significantly degrade the performance of the evaluated instrument channels during normal plant operation. All potential time-related degradation mechanisms have insignificant effects in the time frame of interest (maximum of 30 months). PG&E will continue to perform the maintenance required to maintain the qualification of this safety related equipment.

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

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

The proposed pressurizer water level trip setpoint, RCS flow safety analysis limit, and various allowable value changes provide adequate margin to accommodate instrument channel uncertainty over a 30-month operating cycle. Plant equipment, which will be set at, or more conservative than, the trip setpoints, will provide protective functions to assure that the safety analysis limits are not exceeded. The change to the RCS loss of flow safety analysis limit does not create the possibility of a new or different kind of accident since the setpoint will remain as currently specified and only results in an insignificant delay in the plant response to the accident.

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

For almost all the existing DCPP RTS/ESFAS setpoints, the existing difference between the safety analysis limit and the setpoints was sufficient to accommodate any changes in instrument uncertainty.

The change in the pressurizer water level - high setpoint does not affect a safety analysis limit and, therefore, has no effect on a margin to safety. Since the normal pressurizer level is maintained at 60 percent span and the no-load T_{avg} control level is 22 percent span, a change in the setpoint from less than or equal to 92 percent span to less than or equal to 90 percent span is not significant to either DCPP plant operation or safety.

The change in the RCS loss of flow-low safety analysis limit from 87 percent MMF to 85 percent MMF does not affect the existing plant setpoint and was evaluated to have a negligible effect on the limiting conditions of a partial loss of flow accident, a single RCP locked rotor, or RCP shaft break accident. This safety limit change was also found to have no effect on the DCPP minimum DNBR since the minimum DNBR is associated with the complete loss of flow accident. The complete loss of flow accident was evaluated to the Condition II fault criteria applicable to the partial loss of flow accident evaluation and was acceptable.

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

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment request: December 23, 1996

Description of amendment request: The proposed amendment would allow the use of Vantage Plus fuel.

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

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

The probability of occurrence or the consequences of an accident previously evaluated is not significantly increased. The VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble and instrument tubes, IFMs, [intermediate fuel mixing assemblies] and LPD [low-pressure-drop] mid-grids meet the same fuel assembly and fuel rod design bases as VANTAGE 5 (without IFMs) fuel assemblies in the other fuel regions. In addition, the 10 CFR 50.46 criteria will be applied to the ZIRLO™ clad fuel rods, thimble and instrument tubes, IFM grids, and LPD mid-grids. The use of these fuel assemblies will not result in a change to the proposed Indian Point Unit 3 VANTAGE 5 (without IFMs) transition core design and safety analysis limits. The ZIRLO™ clad material is similar in chemical composition and has similar physical and mechanical properties as that of Zircaloy-4. Thus the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. The ZIRLO™ clad fuel rod improves corrosion resistance and dimensional stability. In addition, the incorporation of LPD mid-grids and IFMs improves dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel assemblies material changes as specified in this report, the radiological consequences of accidents previously evaluated in the safety analyses remain valid. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

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

The possibility for a new or different type of accident from any accident previously evaluated is not created, since the VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble, and instrument tubes, IFMs, and LPD mid-grids will satisfy the same design bases as that used for VANTAGE 5 (w/o IFMs) fuel assemblies in the other fuel regions. Since the original design criteria is being met, the ZIRLO™ clad fuel rods, thimble and instrument tubes, IFMs, and LPD mid-grids will not be an initiator for any new accident. All design and performance criteria will continue to be met and no new single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alterations to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

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

The margin of safety is not significantly reduced, since the VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble and instrument tubes; IFMs, and LPD mid-grids do not change the proposed Indian Point 3 VANTAGE 5 (w/o IFMs) transition core design and safety analysis limits. The use of these fuel assemblies containing fuel rods, thimble and instrument tubes with ZIRLO™ cladding alloy; IFMs and LPD mid-grids will take into consideration the normal core operating conditions allowed in the Technical Specifications. For the transition core and each future cycle reload core, these fuel assemblies will be specifically evaluated using standard reload design methods and approved fuel rod design models and methods. This will include consideration of the core physics analysis, peaking factors and core average linear heat rate effects. In addition, the 10 CFR 50.46 criteria will be applied each cycle to the ZIRLO™ clad fuel rods, thimble and instrument tubes, IFMs, and LPD mid-grids. Analyses or evaluations will be performed each cycle to confirm the 10 CFR 50.46 will be met. Therefore, the margin of safety as defined in the Bases to the Indian Point Unit 3 Technical Specifications and VANTAGE 5 (w/o IFMs) ZIRLO™ licensing amendment approval is not significantly reduced.

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

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

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

NRC Project Director: S. Singh Bajwa, Acting

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

Date of amendment request: July 9, 1996 (TXX-96393)

Brief description of amendments: The proposed changes would increase the minimum allowable value of the Unit 1 Steam Line Pressure--Low Safety Injection and Steam Line Isolation functions. These changes are needed to ensure that the instrumentation error is properly accounted for in the Technical Specifications.

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

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

The changes in the License Amendment Request proposes more restrictive setpoint Allowable Values for the Steam Line pressure--Low channels of the Engineered Safety Features Actuation System (ESFAS). These more restrictive values assure that all applicable safety analysis limits are being met. Changing an Allowable Value in the Technical Specifications has no impact on the probability of occurrence of any accident previously evaluated. None of the accident analyses were affected, therefore, the consequences of all previously evaluated accidents remain unchanged.

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

The proposed changes involve the use of a more conservative value for the Allowable Value for the Steam Line Pressure--Low Safety Injection and Steam Line Isolation functions. As such, none of the changes effect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There were no changes made to any of the accident analyses or safety analysis limits as a result of this proposed change. Further, the proposed change does not affect the acceptance criteria for any analyzed event. ESFAS will remain capable of performing its safety function, and the new requirement will continue to provide adequate assurance of that capability. Making the Allowable Value more restrictive provides increased assurance that the channels will function within the safety analysis limits assumed in the safety analyses. The margin of safety established by the Limiting Conditions for Operation also remains unchanged. Thus there is no effect on the margin of safety.

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: July 10, 1996 (TXX-96405), as supplemented by letter dated October 1, 1996 (TXX-96475)

Brief description of amendments: The proposed change would take credit for the addition of train oriented Fan Coil Units for each UPS & Distribution Room and would provide redundancy to the existing Air Conditioning Units.

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

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

The UPS HVAC System is a support system for other safety related equipment, primarily the Uninterruptible Power Supplies and some of their distribution equipment. The only impact that this system can have on the probability or consequences of an accident must result from the failure of the system to provide adequate support to the supported safety related equipment when that supported safety related equipment is required to operate.

Allowing same train cooling to satisfy the LCO is considered equivalent to the existing Technical Specification. The proposed changes allow the use of the same train UPS Room Fan Coil Units or the same train UPS A/C Train to support a UPS & Distribution Room.

Surveillance requirements are added or modified to ensure that the credited support equipment will be available when needed. Unnecessary starts of the UPS A/C Trains have been eliminated from the specifications. Overall, this is considered an enhancement that will increase the reliability of the UPS HVAC Systems. Because both the existing specification and the proposed revision to the specification continue to ensure normal support and the availability of at least one train of equipment in the event of a design basis accident, with the same or increased reliability, the consequences of an accident previously evaluated is not affected.

Changing the specification from a ''common'' specification which impacts both units simultaneously to a specification which applies to both units separately is basically just an administrative change. Having ''common'' specifications is an aid to the operator to provide an alert that both units are affected. With the new LCO, both units may not be affected because rooms may now be cooled separately. Because both CPSES Units remain properly covered, however, this change will not significantly increase the probability of consequences of an accident.

The revision to the existing ACTION is considered equivalent except for the change of the Allowed Outage Time (AOT) from seven days to 30 days. This change is based on the significance of the heating and cooling function but does represent an increase in AOT and thus an increase in the probability that the supported functions could be unavailable. This increase is not considered significant based on the following several factors:

a)the systems design is based on a conservative assessment of the worst postulated conditions in the rooms;

b) generally, less than design cooling is required and a short duration or partial failure may have little or no impact on the system—s ability to perform its function;

c) the multiple backups available (two UPS A/C Trains and only one UPS Room Fan Coil Unit per each room) increase the potential of restoring additional cooling if needed;

d) the ability to perform alternate actions if normal cooling is lost such as circulating air via existing fans or portable fans thereby extending the time before cooling must be restored; and

e) the extended AOT would allow more time and opportunity to perform corrective maintenance to ensure high equipment reliability.

The new ACTION for loss of cooling reflects requirements that already exist in the Technical Specifications. The AOT for this ACTION statement is 72 hours which is based on the risk from an event occurring requiring the inoperable UPS A/C Train, and the remaining UPS Room Fan Coil Units and A/C Train fans providing the required protection.

The new ACTION for loss of cooling and ventilation reflects a conservative response to the potential impact of such a condition. The proposed AOT is one hour. One hour is based on the time lag available from the operating temperature to the maximum Technical Specification limit of the UPS & Distribution Rooms. The addition of a specific ACTION in lieu of relying on Specification 3.0.3, although essentially equivalent, is consistent with the methodology of the improved Standard Technical Specifications and alerts the operator to the significance of the situation.

The changes made to the surveillance ensure that the UPS Room Fan Coil Units will operate. The UPS Room Fan Coil Units are connected to the emergency busses and TS 4.8.1.1.2f. demonstrates the energization of emergency busses with permanently connected loads. The changes made to the 18 month surveillances on the UPS A/C trains were changed from the Safety Injection signal with the Blackout Test signal to ''... actual or simulated actuation signal''. This is consistent with NUREG-1431, ''Standard Technical Specifications Westinghouse Plants''.

The changes to the BASES are descriptive in nature to reflect the other changes and by themselves have no impact on the probability or consequences of an accident.

The ability to cope with station blackout and design basis fires is maintained or enhanced. For station blackout coping, the UPS A/C fans are considered to remain available while additional cooling is provided by a single available Fan Coil Unit.

In summary, the proposed changes take advantage of the increased reliability offered by the revised system design. It also maintains the level of support provided by the system while at worst, allowing a slight decrease in availability (in certain situations) which is not considered significant. As a result, it is concluded that none of the changes made to the existing Technical Specification involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Revising this specification to take credit for the new UPS Room Fan Coil Units, to take credit for same train UPS A/C Train support for a UPS and Distribution Room, to make the specification unit specific instead of common, to make the surveillances appropriate for the credited equipment, and to make the action statements appropriate for the credited equipment and their significance, does not by itself alter plant hardware. Plant procedures are only altered to the extent that the revised specification will allow different configurations of equipment in the UPS HVAC System to be operated at different times. These changes ensure continued support of the safety related equipment in the affected areas and do not affect the equipment—s failure or failure modes. As a result, these changes to the Technical Specification do not create the possibility of a new or different kind of accident from any previously evaluated.

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

None of the changes being proposed alter the environmental conditions which are to be maintained in the areas supported by an OPERABLE UPS HVAC System during normal operations and following an accident. As a result, the margin of safety for these functions remains the same. The only potential adverse impact is the system's postulated availability, as discussed in the response to question 1 above. This reduction in availability is to a great extent mitigated by the projected increase in system reliability. As noted in the response to question 1, there is no significant impact on the accident analyses. Thus, even if system availability issues were considered an aspect of margin of safety, the proposed changes do not involve a significant reduction in margin of safety.

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: December 17, 1996

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

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

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

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

The piping refurbishment project and the proposed temporary changes to the SW [service water] and CC [component cooling] Technical Specifications have been evaluated to assess their impact on the normal operation of the SW and CC systems and to ensure that the design basis safety functions of each system are preserved. The SW system is required to function during all normal and emergency operating conditions. During normal plant operation, the SW system provides cooling water to the CCHXs, charging pump coolers, instrument air compressor coolers, and control room chiller condensers of both units. Within the first 168 hour Section 3/4.7.4.1.d TS AS [Action Statement] of isolation of the header which is to be repaired, temporary 10'' diameter SW lines (one supply and one return) will be installed to supply the SW to the charging pumps coolers, instrument air compressors coolers, Unit 2 CR chillers and spent fuel pool (SFP) coolers to satisfy design basis conditions. These temporary lines will be routed from the operating part of the 36'' SW headers while the 24'' headers to CCHXs are being repaired. The temporary lines will be dismantled when the repaired header is returned to operation (second 168 hour AS). During the two 35-day periods, one header will operate with its 24-inch piping to/from the CCHXs temporarily blanked. To avoid operation of the SW pump at abnormal conditions (low flow) on this ''partially deadlocked'' header, a temporary cross-connect will be installed to by-pass the CCHXs.

SW system operation with the cross-connect installed was evaluated for design basis accident (DBA) conditions. The DBA condition for the SW system is a loss-of-coolant accident on one unit with simultaneous loss-of-offsite-power to both units. [An] SW system hydraulic analysis has been performed to verify that adequate flow is provided to the containment recirculation spray heat exchangers (RSHXs) with the temporary cross-connect installed and throttled open, assuming the occurrence of the most limiting single failure. Therefore, there is no increase in probability or consequences of the DBA condition.

Utilizing only one SW header to supply flow to the CCHXs has the potential to affect the reliability of the CC system and all of the equipment cooled by CC. A review of the equipment affected by this phase of the SW restoration project was performed to evaluate the impact on initiating event frequency. Since the SW system and CC system are support systems used to remove heat, a failure in either of these systems does not affect the initiating event frequency of any design basis event. Additionally, an estimate of the impact on core damage frequency is provided below. The impact on the North Anna Probabilistic Safety Assessment (PSA) during implementation of this DCP [design change package] is similar to impact of work performed under DCP-94-010 since the scope of work of both DCPs is repair/replacement of different portions of the same 24'' SW headers to CCHXs. The only difference from a PSA standpoint is that CDF [core damage frequency] for DCP-94-010 was calculated based on 140 days supply of CCHXs from one SW header while per this DCP it is only 70 days. Therefore, results of PSA evaluation for DCP-94-010 are conservatively applied to this DCP. The activities to be performed during the refurbishment project and the various system alignments required have been evaluated using the Individual Plant Examination (IPE) Probabilistic Safety Assessment (PSA) model for North Anna Power Station. This model is used in a manner that is generally consistent with the Electric Power Research Institute (EPRI) PSA Applications Guide TR-105396. The effect on the PSA model is a slight increase in the frequency of reactor trips and an increase in the probability of RHR [residual heat removal] failure.

The increased frequency of reactor trips is due to the decreased reliability of the CC system to supply cooling to the RCP [reactor coolant pump] motor. When only one SW header is available to the CC heat exchangers the frequency of losing this single header is dominated by the probability of both SW pumps failing. Also considered was the frequency of pipe rupture anywhere in the single available header. When the single SW header fails to supply cooling to the CC heat exchangers, the CC system will heatup causing inadequate cooling for sustained operation of the RCPs. Tripping these pumps results in a reactor trip. The second SW header can be expected to supply other equipment with cooling. This scenario is appropriately modeled as a reactor trip with main feedwater available initiating event. A sensitivity analysis shows the increase in CDF to be about 1E-8/year. The total effect of this DCP includes a failure analysis of the reactor coolant pump and motor in case of loss of CCW.

The CC system is also included in the PSA model as a support system for RHR cooling. The RHR system is used to reduce reactor coolant system temperatures from 350°F (hot shutdown) to 140°F (cold shutdown). The only accident initiator that requires the unit to be cooled down and placed on RHR cooling are sequences which are initiated with a steam generator tube rupture. (Note that, for the North Anna plant design, RHR is separate from the safety injection system and the low head safety injection pumps.) The increased probability for the loss of RHR when only one SW header is available to the CCHXs is estimated using fault tree analysis and is dominated by the failure of both SW pumps. The probability for the loss of both SW pumps aligned to the CCHXs is estimated to be 1.5E-4. The effect of this increase in RHR failure probability was determined by adding this probability to the top single event in the RHR function and recalculating the new CDF. The resulting increase in CDF as a result of RHR system failure following a steam generator tube rupture is less than 1E-8 per year.

The CC system is further included in the PSA model as part of the loss of RCP seal cooling as an initiating event and as a loss of function during other initiating event scenarios. The effect on the probability for a loss of RCP seal cooling due to losing CC cooling to the RCP thermal barriers is negligible due to the high reliability of the charging system to provide seal injection.

The total effect of this DCP on core damage frequency (CDF) was estimated by a sensitivity analysis combining both the change in the reactor trip initiating event frequency and the increased failure probability of RHR. It was evaluated that during implementation of this DCP, CCHXs will be supplied from one SW header for 70 days (35 x 2=70), therefore, the increase in CDF previously evaluated in DCP-94-010 based on 140 days is conservative. This DCP does not affect the containment systems and there would not be any significant change in off-site dose since the containment heat removal portion of the SW system is not affected and the increase in CDF is insignificant. The small increase in CDF calculated for the repair activities and the procedure developed to provide contingency actions result in the conclusion that this work does not represent a significant increase in core damage frequency.

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

The proposed changes to the allowed outage time only provide operational flexibility needed to perform necessary repairs. During the project, there will be a significant time period when all the CCHXs are aligned to one SW loop. The possibility of an interruption of SW supply to the heat exchangers during a DBA is eliminated by defeating the closure of the 24-inch SW isolation MOVs [motor-operated valves] to the CCHXs on [an] SI/CDA [safety injection/containment depressurization actuation] signal. Both SW headers will be available for equipment required for safe shutdown of the units (i.e., RSHXs, charging pumps, and CR/ESGR [control room/emergency switchgear room] chillers). The SW pipe repair activities and the installation/removal of the SW cross-connect and temporary piping do not create the possibility for a malfunction of equipment different than previously evaluated. Results of the Johnston Pump NPSH [net positive suction head] test proved to be satisfactory for the anticipated SW pump flow rates under modes of station operation for this project, therefore, the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report will not be created. Based on the above, implementation of the restoration project and approval of the proposed Technical Specifications changes will not introduce any new accident initiators nor affect the performance of accident mitigation systems.

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

The proposed changes to the schedule only provide operational flexibility to perform the required SW pipe refurbishment. The Technical Specifications continue to require the SW and CC systems to remain functional during the period with a single SW supply to the CCHXs. As stated in item (1) above, the SW system is fully capable of performing its DBA function during the course of the pipe refurbishment project with the proposed Technical Specification changes in place. The effect of this pipe refurbishment project on CC system reliability was estimated by a sensitivity analysis combining both the change in the reactor trip initiating event frequency and the increased failure probability of RHR resulting in about a 1E-8 per year increase in CDF. Since this project will not affect the containment systems, there would not be any significant change in off-site dose, except that resulting directly from the slight increase in CDF.

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

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: F. Mark Reinhart (Acting)

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.

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

Date of application for amendments: September 10, 1996

Brief description of amendments: The amendments extend the automatic actuation logic channel functional test interval of the Engineering Safety

Features Actuation System and the surveillances test interval of the containment sump isolation valves from monthly to quarterly.

Date of issuance: January 23, 1997

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

Amendment Nos.: 218 and 195

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 9, 1996 (61 FR 52963) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated January 23, 1997 No significant hazards consideration comments received: No

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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: October 31, 1996

Brief description of amendments: The amendments relocate the requirements for seismic monitoring instrumentation from Technical Specification (TS) 3/4.3.7.2, ''Seismic Monitoring Instrumentation'' to licensee-controlled documents in accordance with Generic Letter 95-10, ''Relocation of Selected Technical Specifications Requirements Related to Instrumentation.'' The amendments also add a condition to the operating licenses which approves the relocation of the TS requirements to the UFSAR.

Date of issuance: January 29, 1997 Effective date: Immediately, to be implemented within 90 days.

Amendment Nos.: 117, 102

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

Date of initial notice in Federal Register: December 18, 1996 (61 FR 66703) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 1997. 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 No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: September 9, 1996, as supplemented December 20, 1996

Brief description of amendment: The amendment revises the Minimum Channels Operable requirement of Item 4.c (Steam Line Isolation, Containment Pressure Intermediate--High-High) of Technical Specification (TS) Table 3.3-3 from 3 channels to 2 channels provided the provisions of Action Statement 14 are followed. This change makes this Beaver Valley Power Station, Unit No. 1 TS consistent with the comparable Beaver Valley Power Station, Unit No. 2 TS. The amendment also revises the minimum charging pump discharge pressure in TS 3/4.5.5 and associated Bases from 2311 psig to 2397 psig. This change ensures that safety analysis assumptions for safety injection flow are met.

Date of issuance: January 27, 1997

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

Amendment No: 201

Facility Operating License No. DPR-66. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55032) The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or the original notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1997. No significant hazards consideration comments received: No

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

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of application for amendment: August 15, 1996, and as supplemented by letters dated October 28, November 15, 1996, and January 7, 1997.

Brief description of amendment: The amendment changes the Clinton Power Station (CPS) Technical Specifications to incorporate the revised Safety Limit Minimum Critical Power Ratio (SLMCPR) as calculated by General Electric (GE) for CPS Cycle 7. The need to change the SLMCPR resulted from the 10 CFR Part 21 condition reported by GE in their letter to the NRC dated May 24, 1996.

Date of issuance: January 22, 1997

Effective date: January 22, 1997

Amendment No.: 113

Facility Operating License No. NPF-62: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47978). The licensee's letters of October 28, November 15, 1996, and January 7, 1997, provided clarifying information and did not make significant changes to the initial Federal Register notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 22, 1997. No significant hazards consideration comments received: No

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Northeast Nuclear Energy Company, Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: August 29, 1996

Brief description of amendment: The amendment revises the Technical Specifications to (1) modify the applicability requirements for certain radiation monitors so that the radiation monitors are required to be operable only when secondary containment integrity is required to be operable; (2) delineate when secondary containment integrity is required; (3) modify standby gas treatment operability requirements; (4) make editorial corrections to clarify the configuration of the radiation monitors; and (5) revise the associated Bases sections.

Date of issuance: January 14, 1997

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

Amendment No.: 98

Facility Operating License No. DPR-21: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 17, 1996 (61 FR 54242) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 14, 1997 No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 28, 1996, as supplemented by letters dated November 4 and 5, and December 9, 1996

Brief description of amendments: These amendments revise the technical specifications to incorporate performance-based testing, in accordance with 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors,'' Option B. This option allows utilities to extend the frequencies of the Type A Containment Leak Rate Test, and Type B and C Local Leak Rate Tests based on the performance and design of the containment and components.

Date of issuance: January 24, 1997

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

Amendment Nos.: 118 and 81

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55038) The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or the original notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated Janaury 24, 1997. No significant hazards consideration comments received: No

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

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

Date of application for amendments: May 20, 1996

Brief description of amendments: These amendments revise Technical Specifications (TS) Sections 3/4.4.9.2, 3/4.9.11.1, 3/4.9.11.2, and the associated TS Bases 3/4.4.9 and 3/4.9.11, to more clearly describe that the Residual Heat Removal (RHR) system Shutdown Cooling mode of operation consists of four ''subsystems.'' These TS sections pertain to plant operations during Operational Conditions (OPCONs) 4, ''Cold Shutdown'' and 5, ''Refueling.'' In addition, the proposed TS change would make administrative changes to TS Section 3/4.4.9.1 to ensure consistency in terminology regarding the description of Shutdown Cooling ''subsystems.'' The proposed TS changes are consistent with the guidance delineated in the Improved TS (i.e., NUREG-1433, Revision 1, ''Standard Technical Specifications General Electric Plants, BWR/4,'' dated April 1995) which indicates that the RHR Shutdown Cooling mode of operation is comprised of two loops and four subsystems (i.e., two subsystems per loop).

Date of issuance: January 28, 1997

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

Amendment Nos.: 119 and 82

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55036) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 28, 1997. No significant hazards consideration comments received: No

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

Philadelphia Electric Company, Docket No. 50-353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: August 5, 1996, as supplemented December 4, 1996

Brief description of amendment: The amendment revised TS Section 2.1 and its associated TS Bases to reflect the change in the Minimum Critical Power Ratio Safety Limit due to the plant specific evaluation performed by General Electric Company (GE), for Limerick Generating Station, Unit 2, Cycle 4.

Date of issuance: January 29, 1997

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

Amendment No.: 83

Facility Operating License No. NPF-85. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 6, 1996 (61 FR 57491) The December 4, 1996, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or the initial notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 29, 1997. No significant hazards consideration comments received: No

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

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

Date of application for amendments: March 29, 1996, as supplemented by letters dated December 5, 1996, and January 15, 1997

Brief description of amendments: These amendments modify Technical Specification (TS) Section 4.5.1.d.2.b to delete the requirement to perform in-situ functional testing of the Automatic Depressurization System (ADS) valves once every 24-months as part of start-up testing activities.

Date of issuance: January 29, 1997

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

Amendment Nos.: 120 and 84

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: November 6, 1996 (61 FR 57488) The December 5, 1996, and January 15, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the initial notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 1997. 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: October 1, 1996

Brief description of amendment: The amendment allows for a one-time extension of the surveillance intervals for the containment isolation valve seat leakage test, the isolation valve seal water test, the boron injection tank leakage test, the containment spray nozzle test, and the city water backup to the auxiliary boiler feed pump test.

Date of issuance: January 28, 1997

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

Amendment No.: 172

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

Date of initial notice in Federal Register: December 4, 1996 (61 FR

64393) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 28, 1997 No significant hazards consideration comments received: No

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

Public Service Electric & Gas Company, Docket 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 1, 1996, supplemented October 31, 1996

Brief description of amendments: The amendments change Technical Specifications 3/4.7.1.5, ''Main Steam Line Isolation Valves (MSIVs),'' and 3/4.3.2, ''Engineered Safety Feature Actuation System Instrumentation.'' The amendments accommodate entry into Modes 3 and 2 prior to performing MSIV closure time testing in Mode 2, allow additional time for the repair and testing of inoperable MSIVs in certain operating Modes, delete footnotes that are no longer applicable, and change the low steam line pressure trip setpoint value for safety injection, turbine trip and feedwater isolation to make it consistent with the actual plant configuration.

Date of issuance: January 17, 1997

Effective date: Both units, as of date of issuance, to be implemented prior to entry into Mode 3 from the current outage.

Amendment Nos. 187 and 170

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

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55040) The supplemental letter changed the TSs to provide greater consistency with requirements of NUREG-1431 ''Standard Technical Specifications - Westinghouse Plants,'' Revision 1, and did not change the initial proposed no significant hazards consideration determination or the Federal Register notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 17, 1997. 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 24, 1996, as supplemented December 23, 1996

Brief description of amendments: The amendments changed Technical Specification 3/4.7.1.2, ''Auxiliary Feedwater System.'' The changes revised the 18-month surveillance performed on the system's pumps and valves because testing of the turbine driven Auxiliary Feedwater pump can only be performed in higher modes when there is sufficient secondary steam pressure.

Date of issuance: January 23, 1997

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

Amendment Nos. 188 and 171

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

Date of initial notice in Federal Register: November 19, 1996 (61 FR 58905) The December 23, 1996, letter proposed changes to TS 3/4.3.2 to provide consistency with those proposed in the October 24, 1996, letter and therefore did not change the initial proposed no significant hazards consideration determination and was within the scope of the initial notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 23, 1997. 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: September 25, 1996

Brief description of amendments: The amendments relocate the list of containment isolation valves from the Technical Specifications to the Salem Updated Final Safety Analysis Report and correct references. Date of issuance: January 30, 1997

Effective date: Both units, as of date of issuance, to be implemented within 60 days.

Amendment Nos. 189 and 172

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

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55039) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 30, 1997. No significant hazards consideration comments received: No

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

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

Date of application for amendments: January 31, 1996, as revised November 26, 1996. The November 26, 1996, submittal withdrew the proposed change to surveillance tests being performed at power.

Brief description of amendments: These amendments will revise the minimum emergency diesel generator day tank fuel oil volume.

Date of issuance: January 17, 1997

Effective date: January 17, 1997

Amendment Nos.: 203 and 184

Facility Operating License Nos. NPF-4 and NPF-7. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7559) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 17, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 8, 1996, as supplemented August 15, December 2 and December 19, 1996, and January 6, 1997

Brief description of amendments: These amendments revise Technical Specification (TS) Section 15.3.10, ''Control Rod and Power Distribution Limits,'' to improve the clarity of this section and add surveillance requirements to Section 15.4.1, ''Operational Safety Review.''

Date of issuance: January 16, 1997

Effective date: January 16, 1997, with full implementation within 45 days

Amendment Nos.: Unit 1 - 171, Unit 2 - 175

Facility Operating License Nos. DPR-24 and DPR-27: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 13, 1996 (61 FR 10398) The August 15, December 2 and

December 19, 1996, and January 6, 1997, letters provided clarifying information and updated TS pages that were within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 16, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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 and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point, Units 3 and 4, Dade County, Florida

Date of amendment request: December 17, 1996

Description of amendment request: The proposed amendments would modify the Turkey Point Units 3 and 4 Technical Specifications to change the Reactor Coolant Pump (RCP) flywheel surveillance requirement. The proposed change will require RCP flywheel inspections once every ten years.

Date of publication of individual notice in Federal Register: January 10, 1997 (62 FR 1476)

Expiration date of individual notice: February 10, 1997

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

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: December 13, 1996

Description of amendment request: The proposed amendment would approve transfer of Soyland Power Cooperative's 13.21% minority ownership interest in the Clinton Power Station to Illinois Power Company. This action would result in Illinois Power Company becoming the sole owner of the Clinton Power Station.

Date of publication of individual notice in Federal Register: January 29, 1997 (62 FR 4337).

Expiration date of individual notice: February 28, 1997

Local Public Document Room location: : Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

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

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

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

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental

Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By March 14, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, and to the attorney for the licensee.

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

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: January 13, 1997, as resubmitted January 17, 1997, and supplemented January 22, 1997.

Brief description of amendments: The proposed amendments would: evaluate the Unreviewed Safety Question (USQ) associated with the operation of Dresden, Units 2 and 3, with the recently discovered error in the head loss across the Emergency Core Cooling System (ECCS) suction strainers; change the Technical Specification (TS) values by lowering the allowable water temperature in the suppression chamber and ultimate heat sink; change the basis of the TS to allow credit for two psig of containment pressure to compensate for a slight increase in the amount of Net Positive Suction Head (NPSH) deficiency during the first 10 minutes following a design basis accident (DBA); and add a license condition to allow the licensee to change the Updated Final Safety Analysis Report to reflect the use of two psig of containment pressure to compensate for the deficiency in NPSH during the first 10 minutes following a DBA.

Date of Issuance: January 28, 1997 Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 152/147

Facility Operating License Nos. DPR-19 and DPR-25. The amendments revised the Technical Specifications and the Operating Licenses. Press release issued requesting comments as to proposed no significant hazards consideration: Yes January 25, 1997 Joliet Herald News Comments received: No. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Illinois and determination of no significant hazards consideration are contained in a Safety Evaluation dated January 28, 1997.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690

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

NRC Project Director: Robert A. Capra

Dated at Rockville, Maryland, this 5th day of February, 1997.

For the Nuclear Regulatory Commission

Jack W. Roe,

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

[Doc. 97-3324 Filed 2-11-97; 8:45 am]

BILLING CODE 7590-01-F

PHYSICIAN PAYMENT REVIEW COMMISSION

Commission Meeting

AGENCY: Physician Payment Review Commission.

ACTION: Notice of meeting.

SUMMARY: The Commission will hold its next public meeting on Thursday, February 20, 1997 and Friday, February 21, 1997, at the Washington Marriott, 1221 22nd Street NW, Washington, D.C., in the DuPont Salon. The meetings are tentatively scheduled to begin at 10:00 a.m. on February 20 and at 9:00 a.m. on February 21.

In preparation for its March 31, 1997 Annual Report to Congress, the Commission will review the conclusions and recommendations to be contained in chapters on the following topics:

Context for Reform
The Medicare Risk-Contracting Program: Plan Participation and Enrollment
Revising the Method for Determining Medicare Capitation Payments
Implementing Risk Adjustment in the Medicare Program
Promoting Access to Care for Vulnerable Populations in Medicare Managed Care
Access to Care in Medicare Risk Plans
Using Quality and Performance Measures in Medicare
Health Plan Data Needs and Capabilities
Competitive Premium Contribution Models: Options for Medicare
Provider-Sponsored Organizations
Consumer Protection Initiatives for Managed Care
Constraining Spending in Medicare Fee for Service
Improving the Efficiency of Medicare Fee for Service Through Preferred Providers
Medicare Fee Schedule Payment Issues (Work Values, Practice Expense, GPCIs, Impact on Payments)
Access and Beneficiary Financial Liability Under the Medicare Fee Schedule
Secondary Insurance for Medicare Beneficiaries
The Changing Labor Market for Physicians
Academic Medical Centers and the Changing Health Care Marketplace
Payments from a Teaching Hospital and Graduate Medical Education Trust Fund
Managing Health Care for Dually Eligible Beneficiaries
Medicaid: Spending Trends and the Move to Managed Care

Final agendas will be mailed on February 14, 1997 and will be available on the Commission's web site (WWW.PPRC.GOV) at that time.

ADDRESSES: 2120 L Street, N.W.; Suite 200; Washington, D.C. 20037. The telephone number is 202/653–7220.

FOR FURTHER INFORMATION CONTACT: Debbie Kramer, Executive Assistant, at 202/653–7220.

SUPPLEMENTARY INFORMATION: If you are not on the Commission mailing list and wish to receive an agenda, please call 202/653–7220 after February 14, 1997.

Lauren LeRoy,

Executive Director.

[FR Doc. 97–3480 Filed 2–11–97; 8:45 am]

BILLING CODE 6820–SE–M

RAILROAD RETIREMENT BOARD

Agency Forms Submitted for OMB Review

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35), the Railroad Retirement Board has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

SUMMARY OF PROPOSAL(S):

(1) *Collection title:* Employer's Quarterly Report of Contributions Under the RUIA.

(2) *Form(s) submitted:* DC–1.

(3) *OMB Number:* 3220–0012.

(4) *Expiration date of current OMB clearance:* March 31, 1997.

(5) *Type of request:* Extension of a currently approved collection.

(6) *Respondents:* Businesses or other for profit.

(7) *Estimated annual number of respondents:* 550.

(8) *Total annual responses:* 2,200.

(9) *Total annual reporting hours:* 917.

(10) *Collection description:* Railroad employers are required to make contributions to the Railroad Unemployment Insurance fund quarterly or annually equal to a percentage of the creditable compensation paid to each employee. The information furnished on the report accompanying the remittance is used to determine the correctness of the amount paid.

ADDITIONAL INFORMATION OR COMMENTS: Copies of the form and supporting documents can be obtained from Chuck Mierzwa, the agency clearance officer (312–751–3363). Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Board, 844 North Rush Street, Chicago, Illinois 60611–2092 and the OMB reviewer, Laura Oliven (202–395–7316), Office of Management and Budget, Room 10230, New Executive Office Building, Washington, D.C. 20503.

Chuck Mierzwa,

Clearance Officer.

[FR Doc. 97–3494 Filed 2–11–97; 8:45 am]

BILLING CODE 7905–01–M

Agency Forms Submitted for OMB Review

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35), the Railroad Retirement Board has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

SUMMARY OF PROPOSAL(S):

(1) *Collection title:* Nonresident Questionnaire.

(2) *Form(s) submitted:* RRB–1001.

(3) *OMB Number:* 3220–0145.

(4) *Expiration date of current OMB clearance:* March 31, 1997.

(5) *Type of request:* Extension of a currently approved collection.

(6) *Respondents:* Individuals or households.

(7) *Estimated annual number of respondents:* 1,700.

(8) *Total annual responses:* 1,700.

(9) *Total annual reporting hours:* 108.

(10) *Collection description:* Under the Railroad Retirement Act, the benefits payable to an annuitant living outside the United States may be subject to withholding under Public Laws 98–21