1997, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland, has been rescheduled for February 6–8, 1997. The meeting will begin at 8:30 a.m. on Thursday, February 6, 1997, instead of 1:00 p.m. on Wednesday, February 5, 1997. The discussion of the item on ''Design-bases Verification'' scheduled for Wednesday, February 5, 1997, has been postponed to a future meeting as requested by the NRC staff. All other items pertaining to this meeting remain the same as published in the Federal Register on Thursday, January 23, 1997 (62 FR 3539).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Chief, Nuclear Reactors Branch (telephone 301/415–7364), between 7:30 A.M. and 4:15 P.M. EST.

Dated: January 23, 1997.

Andrew L. Bates,

*Advisory Committee Management Officer.*

[FR Doc. 97–2165 Filed 1–28–97; 8:45 am]

**BILLING CODE 7590–01–P**

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## 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 4, 1997, through January 16, 1997. The last biweekly notice was published on January 15, 1997 (62 FR 2185).

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 February 28, 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. 2

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

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

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

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

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

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

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

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

*Date of amendment request:* December 30, 1996

*Description of amendment request:* The amendment revises (1) chemistry data (nickel content) shown on Technical Specification (TS) Figures 3.4-2 and 3.4-3 for TS 3/4.4.9, ''Pressure/Temperature Limits,'' and (2) the associated Bases 3/4.4.9 to reflect changes to chemistry and material properties and changes to comply with recent U.S. Nuclear Regulatory Commission (NRC) rule changes to 10 CFR 50, Appendix G.

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

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

There are no physical changes to any plant equipment created by the proposed changes. The chemistry and material property changes do not impact the ability of the reactor vessel to maintain [its] pressure boundary integrity as previously evaluated. The decrease in EOL USE [End-of-Life Upper Shelf Energy] for weld heat 5P6771 is relatively minor and remains above the required value that has been prescribed by the NRC to provide the necessary level of ductility assumed for reactor vessel integrity evaluations. Therefore, the accident initiating and mitigating aspects of the pressure vessel are not affected. In addition, neither the proposed change requiring the ISLH [In-Service Leak and Hydrostatic] test to be complete before the core is critical nor the proposed change allowing fuel in the reactor vessel during ISLH affects any accident initiating mechanisms. The proposed change requiring the ISLH test to be completed before the core is critical will not increase the consequences of previously evaluated accidents because it conservatively assures the core is subcritical. Although the proposed change allows fuel in the vessel during ISLH utilizing the ISLH Pressure-Temperature (P-T) limits, the consequences of a pressure boundary leak have not changed because ISLH testing is already allowed using the normal plant P-T limits. In addition, the ISLH will be required to be completed before the core is allowed to go critical. The consequences of a leak with fuel in the vessel during ISLH are the same using either the normal P-T limits or the ISLH limits.

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

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

There are no physical changes to any plant equipment or new components created by the proposed changes. The chemistry and material property changes do not impact the pressure boundary integrity of the reactor vessel. The decrease in EOL USE for weld heat 5P6771 is relatively minor and remains above the required value that has been prescribed by the NRC to provide the necessary level of ductility assumed for reactor vessel integrity evaluations. Therefore, the accident initiating aspects of the pressure vessel are not affected. In addition, neither the proposed change requiring the ISLH test to be complete before the core is critical nor the proposed change allowing fuel in the reactor vessel during ISLH creates any new accident initiating mechanisms.

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

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

The changes in chemical and material properties do not adversely affect any reactor vessel integrity evaluations, such as PTS [Pressurized Thermal Shock] or P-T limits. The USE for weld heat 5P6771 does decrease slightly as described in TS Bases Table B 3/4.4-1. However, the predicted EOL USE remains above the value prescribed in 10 CFR 50, Appendix G and is not a significant reduction in the margin of safety. With regard to the proposed changes allowing fuel in the reactor vessel during ISLH, the existing TS Bases specifically state that fuel is not to be in the reactor vessel when the ISLH P-T curve is utilized. However, this change is consistent with the revised 10 CFR 50, Appendix G rule and as such, is not a significant reduction in the margin of safety.

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

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

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

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

*NRC Project Director:* Mark Reinhart, Acting

Commonwealth Edison Company, Docket No. 50-010, Dresden Nuclear Generating Station, Unit 1, Grundy County, Illinois

*Date of amendment request:* October 23, 1996

*Description of amendment request:* The proposed change would amend the Dresden Unit 1 Appendix A Technical Specifications (TS). The proposed amendment is a complete revision of the TS to the same format as Dresden Unit 2/3 TS.

*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. Will operation of the facility according to this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. In general the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis (Decommissioning Plan). Implementation of these changes will not reduce reliability of equipment assumed to operate in the current safety analysis (Decommissioning Plan), or will provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden Station Unit 1's Technical Specifications in general is based on STS [Standard Technical Specifications] guidelines or NRC accepted changes to other facilities such as Trojan or San Onofre Unit 1. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden Station Unit 1. The proposed amendment is consistent with the current safety analysis (Decommissioning Plan) and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis (Decommissioning Plan), or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

2. Will operation of the facility according to this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated.

No. In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis (Decommissioning Plan). Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis (Decommissioning Plan).

The proposed amendment for Dresden Station Unit 1's Technical Specifications in general is based on STS guidelines or NRC accepted changes to other facilities such as Trojan or San Onofre Unit 1. The proposed amendment has been reviewed for acceptability at the Dresden Nuclear Power Station considering similarity of system or component design versus the STS of later operating plants. Any deviations from STS requirements do not create the possibility of a new of different kind of accident previously evaluated for Dresden Station, Unit 1. No new modes of operation are introduced by the proposed changes. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis (Decommissioning Plan). Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the later individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of human performance assumed in the safety analysis (Decommissioning Plan), or provide enhanced assurance that specified parameters remain within their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 6.0 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden Station. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

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

*NRC Project Director:* Seymour H. Weiss

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

*Date of amendment request:* December 2, 1996

*Description of amendment request:* The proposed amendments would revise Technical Specification 3/4.4.2 to reduce the number of required Safety/Relief Valves (SRVs). This change will support a modification to remove five of the currently installed SRVs due to the current excess capacity, and to reduce maintenance costs and worker radiation dose. The current requirement for 17 of the 18 installed SRVs to be operable would be changed to require 12 of the 13 installed SRVs to be operable.

*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:

The probability of an accident previously evaluated will not increase as a result of this change, because the change in valve configuration, and the accompanying piping modification does not alter any of the initiators of an accident or cause them to occur more frequently. The piping modifications will be performed consistent with the current piping classifications for the affected components. Removal of the SRVs will not impact the ability of the remaining SRVs to perform their functions, as described below.

The consequences of an ASME Overpressurization Event are not significantly increased and do not exceed the previously accepted licensing criteria for this event. General Electric (GE) has calculated the revised peak vessel pressure for LaSalle Station to be 1341 psig, which is below the 1375 psig criterion of the ASME Code for upset conditions, referenced in Section 5.2.2, Overpressurization Protection, of the Updated Final Safety Analysis Report (UFSAR), and NUREG-0519 (Safety Evaluation Report related to the operation of LaSalle County Station, Units 1 and 2, March 1981), and Section 15.2-4, Closure of Main Steam Isolation Valves (BWR) of NUREG-0800 (Standard Review Plan). The consequences of this event will continue to be verified on a cycle-specific basis, beginning with LaSalle Unit 1 Cycle 9 (L1C9). These analysis results will be approved as part of the normal reload licensing 10 CFR 50.59 processes.

GE has also performed an analysis of the limiting Anticipated Transient Without Scram (ATWS) event, which is the MSIV Closure Event (MSIVC). This analysis calculated the peak vessel pressure to be 1378 psig, which is well below the 1500 psig criterion of the ASME Code for emergency conditions. General Electric has verified that these results will not be impacted with the introduction of Siemens fuel.

The conclusions given in the safety analyses with regards to primary containment dynamic loads, main steam piping loads, Loss-of-Coolant Accident (LOCA) impact, Minimum Critical Power Ratio (MCPR) impact and SRV availability also show that current accident and transient analyses are not impacted by this change beyond those reanalyzed by GE and discussed above.

There is no increase in the amount or types of radioactive release for any of the affected accidents or transients.

Therefore, there is not a significant increase in the consequences of an accident previously evaluated.

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

The as-left SRV piping configuration will continue to be consistent with the current classifications for these piping and supports, and have been evaluated by Sargent and Lundy analyses. This ensures no different types of events may be caused by piping failures at these locations. This is the only physical modification proposed by this submittal, and it will not create the possibility of a new or different kind of accident from those previously evaluated. Other systems are not modified with this change and have been shown in this submittal to continue to function as intended with the new system configuration, with the exception of the abandoned discharge line snubbers which may be replaced with struts, except where they will be retained as snubbers due to thermal expansion requirements. The changed supports are required to function only as struts with the revised piping. Consideration and evaluation of this function ensure no new or different accidents are created.

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

While the calculated peak vessel pressures for the ASME Overpressurization Event and the MSIVC ATWS Event are increased due to the proposed SRV removals, the new peak pressures remain below the respective licensing acceptance limits associated with these events.

The actual cycle-specific reload analysis of the ASME Overpressurization Event will be verified to be within the licensing acceptance limit for that event prior to each cycle startup, as required in the normal reload 10CFR50.59 process. These licensing acceptance limits have been previously evaluated as providing a sufficient margin of safety. For other accidents and transients, including suppression pool loadings, the SRV removals have a negligible, if any, effect on the results, so the margin of safety is preserved.

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

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

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

*NRC Project Director:* Robert A. Capra

### Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

*Date of amendment request:* August 14, 1996

*Description of amendment request:* The proposed amendment would revise Technical Specification Sections 3.3 (Engineered Safety Features) and 6.9.1.9 (Core Operating Limits Report (COLR)); the basis of Section 3.3, 3.6 (Containment) and 3.10 (Control Rods). These changes would incorporate the best estimate approach into the licensing basis for the Indian Point Unit No. 2 large break loss-of-coolant accident (LOCA) analysis.

*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 evaluated?

Response:

No physical changes are being made by this change. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a loss-of-coolant accident. The consequences of a LOCA are not being increased. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 50.46 paragraph b, that is it meets the five criteria listed in Section II [see application dated August 14, 1996] of this evaluation. No other accident is potentially affected by this change. Therefore, neither the probability nor the consequences of an accident previously analyzed is increased due to the proposed change.

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

Response:

There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of existing plant equipment. All plant systems will perform equally during the response to a potential accident. Therefore, the possibility of a new or different kind of accident than previously analyzed will not be increased.

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

Response:

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the Indian Point Unit No. 2 reactor system during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties have been analyzed to provide assurance that the most severe postulated loss of coolant accidents were calculated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46 paragraph b) are met. Therefore the proposed amendment 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 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 10610.

*Attorney for licensee:* Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

*NRC Project Director:* S. Singh Bajwa, Acting Director

Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

*Date of amendment request:* August 21, 1996

*Description of amendment request:* The proposed amendment would change the licensee's Technical Specifications (TSs) Section 3.3.G (Hydrogen Recombiner System and Post-Accident Containment Venting System), the basis for Section 3.3.G, and Section 4.4, Table 4.4-1 (Containment Isolation Valves). The change would remove the existing flame-type hydrogen recombiners, its supporting equipment, and replace it with passive autocatalytic recombiners (PARs). In addition, the design basis analysis of post-accident hydrogen generation would be recalculated.

*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 evaluated?

Neither the probability nor the consequences of a post-LOCA [loss-of-coolant accident] combustible gas accident are increased by the change in recombiners or in the change to hydrogen generation analysis. The probability of a 10 CFR 59.44 type LOCA is not affected. The consequences of such an accident are not significantly changed.

Accidents associated with failure of the flame-recombiner flue (hydrogen/oxygen) system as well as with failure of the flame-recombiner containment isolation valves have been eliminated.

No other accident is potentially affected by this change.

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

No new modes of plant operation are being introduced other than elimination of operation of the flame-type recombiners and associated support equipment. Recombiner failure is believed to be far less likely with the PAR design but in any event, the containment vent system is being maintained in its current role as backup to recombiner systems. All other plant systems will perform equally during the response to a potential accident. Therefore, the possibility of a new or different kind of accident than previously analyzed will not be increased.

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

The proposed amendment involves margin in the hydrogen flammability limit, in the hydrogen generation assumptions and in the number of PAR devices assumed. Furthermore, sensitivity analysis on PAR effectiveness indicates that additional margin exists for success even with degraded PAR performance. It has been shown by the analysis that the criteria of 10 CFR 50.44(d) can be met with margin. Therefore, the proposed amendment 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 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 10610.

*Attorney for licensee:* Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

*NRC Project Director:* S. Singh Bajwa, Acting Director

Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

*Date of amendment request:* August 22, 1996

*Description of amendment request:* The proposed amendment would revise the licensee's Technical Specification Sections 3.3 and 4.5 (Engineered Safety Features). The proposed revision would delete the requirement to utilize sodium hydroxide (NaOH) as an additive in the posted-accident containment spray system.

*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:

...consistent with the Commission's criteria in 10 CFR 50.92, we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revisions are based on conservative analyses utilizing new, approved methodologies. The analysis shows the sodium hydroxide spray additive can be removed without significantly affecting the radiological consequences of a postulated LOCA [loss-of-coolant accident] and that the calculated off-site doses would remain within the 10 CFR 100 guidelines. In order to maintain acceptable pH levels in the recirculating ECC [emergency core cooling] solution, baskets of trisodium phosphate will be stored in strategic locations in containment.

2) create the probability of a new or different kind of accident from any accident previously evaluated. The proposed change allows the containment safeguards to mitigate the consequences of a design basis LOCA in a manner equivalent to that previously approved.

3) involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met and remain conservative.

Therefore, based on the above, we conclude that the proposed changes do not constitute a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 10610.

*Attorney for licensee:* Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

*NRC Project Director:* S. Singh Bajwa, Acting Director

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

*Date of amendment request:* January 3, 1997

*Description of amendment request:* The proposed amendments would eliminate from various parts of the Technical Specifications any requirement for the low steam pressure signal as an initiator of safety injection. The licensee stated that the function of the signal is adequately performed by other signals (such as the low pressurizer pressure signal).

*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.

The proposed change, to delete the SI [safety injection] signal on low steam line pressure, will only prevent an unnecessary SI actuation as an event occurs which involves secondary system depressurization. No consequences will significantly increase, because for each event previously analyzed it has been shown that either SI on low steam pressure is not demanded, or that another SI signal (e.g., low pressurizer pressure) is generated in sufficient time to meet applicable acceptance criteria. The probability of an accident will not increase.

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

The initiation of SI on a low steam line pressure signal may occur during events which involve a depressurization of the secondary side, including excessive auxiliary feedwater addition. There are other SI initiation signals which will accomplish this same function if needed. Removing this actuation signal will not create any new failure modes or necessitate any new hardware configurations (other than the deletion of the signal itself). No new accident scenarios are created.

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

Analysis has shown that for any transient for which SI would have occurred on low steam line pressure, transient response is maintained within acceptable limits. Steam line break mass and energy releases inside containment do not violate the existing environmental qualification envelope. Steam line breaks outside containment are not adversely affected by this change.

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

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

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

*NRC Project Director:* Herbert N. Berkow

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit Nos. 1 (ANO-1), Pope County, Arkansas

*Date of amendment request:* November 26, 1996

*Description of amendment request:* Change Reactor Coolant System Pressure and Temperature Curvers

*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:

Criterion 1 - Does not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change revises the pressure/temperature limits in accordance with the 10 CFR 50.60 requirements or in accordance with Code Case N-514. This approach utilizes the latest NRC guidelines relative to estimating neutron irradiation damage of the reactor vessel, as well as maintaining conservative limits with respect to the low temperature overpressure protection (LTOP) system. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change will not create the possibility of a new or different kind of accident from any previously evaluated since it does not introduce new systems, failure modes or plant perturbations. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change will not involve a significant reduction in the margin of safety since the proposed pressure/temperature limitations have been developed consistent with the requirements of 10 CFR 50.60. The operational limits have been developed to maintain the necessary margins of safety through 32 effective full power years using methodologies previously reviewed and approved by the NRC. The objective of these limits is to prevent non-ductile failure during any normal operating condition, including anticipated operational occurrences and system hydrostatic tests.

The LTOP safety factors are based on reanalyzed conditions for 32 effective full power years of operation utilizing methodology contained in ASME Code Case N-514. The LTOP evaluation under Code Case N-514 for low temperature transients is considered more appropriate than the ASME Section XI. The code case establishes a factor of 110% of the pressure determined to satisfy Appendix G, paragraph G-2215 of ASME Section XI, Division 1 as a design limit, instead of 100% required by Section XI. This proposed alternative is acceptable because the Code Case recognizes the conservatism of the ASME Appendix G curves and allows establishing a LTOP setpoint which retains an acceptable margin of safety while maintaining operational margins for reactor coolant pump operation at low temperatures and pressures. The Code Case provides an acceptable margin of safety against flaw initiation and reactor vessel failure, and reduces the potential for an undesired LTOP actuation. The application of Code Case N-514 for ANO-1 will ensure an acceptable level of safety. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

*NRC Project Director:* William D. Beckner

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

*Date of amendment request:* October 7, 1996

*Description of amendment request:* Modify Plant Protection System Test Interval to 123 days.

*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:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed changes included in this amendment request are being made to surveillance intervals, allowances to use CISAM elements and various administrative changes. These changes do not alter the functional characteristics of any plant component and do not allow any new modes of operation of any components. These changes do not involve a significant increase in the probability of any event initiator to occur. Therefore, this amendment request does not involve a significant increase in the probability of any accident previously evaluated.

Increasing the surveillance interval for the RPS and ESFAS instrumentation has two principal effects with opposing impacts on risk. The first impact is a slight increase in core damage frequency that results from the increased unavailability of the instrumentation in question from the extended testing interval. The unavailability of the tested instrumentation components is translated to result in a failure of the reactor to trip, an anticipated transient without a scram, or a failure of the appropriate engineered safety feature to actuate when required. The opposing impact on risk is the corresponding reduction in core damage frequency that would result due to the reduced exposure of the plant to test induced transients.

Representative fault tree models were developed for ANO-2 and the corresponding core damage frequency increases and decreases were quantified in CEN-327 and CEN-327 Supplement 1. The NRC staff found that changes in the RPS unavailabilities that result from extending the surveillance test interval (STI) from 30 days to 90 days were not considered to be significant. Estimates of the reduction in scram frequency from the reduction in test induced scrams and the corresponding reduction in core damage frequency were found acceptable. Sequential testing intervals of 90 days were found to result in a net reduction in risk.

CE NPSD-576 employed the same methodology used in CEN-327 and its supplement to evaluate the impact of extending the surveillance intervals from monthly sequential testing to every four months (triannual) on a staggered test basis. The corresponding changes in RPS and ESFAS unavailabilities are quantified in CE NPSD-576 and are shown to be less than their counterparts in CEN-327 and its supplement. Thus, triannual staggered testing should be acceptable as it results in lower RPS and ESFAS unavailabilities than for a 90 day test interval with sequential testing which has been found to be acceptable to the NRC.

The TS amendment request provided the option to use cycle independent shape annealing matrix (CISAM) elements. The CISAM elements will be validated during startup testing and will be required to meet additional acceptance criteria as well as that used for the 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 CPCs. Therefore, the CPCs will operate as designed and this change will not affect the consequences of any accident previously evaluated.

The CPC addressable constant surveillance requirements and the various administrative changes affected by this TS change do not affect the consequences of any accident previously evaluated.

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

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

This amendment request does not involve any changes in equipment and will not alter the manner in which the plant will be operated.

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

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

The RPS/ESFAS extended testing interval yields no significant reduction in the margin to safety. The instrument drift occurring over the proposed STI will not cause the setpoint values to exceed those assumed in the safety analysis and specified in the TS. There are no changes to equipment or plant operations that will result from this change. The implementation of these proposed changes is expected to result in an overall improvement in safety due to the fact that reduced testing will result in fewer inadvertent trips, less frequent actuation of EFAS components, and less frequent distraction of the operations personnel.

The CPC addressable constant surveillance interval extension included in this amendment request is consistent with the methodology found in NUREG-1432, ‘‘Standard Technical Specifications Combustion Engineering Plants’’ (ISTS). Requiring the addressable constant verification to be performed as part of the CPC channel functional test should detect an error in these constants prior to restoring the channel to operable status instead of allowing the error to go undetected until the next surveillance period. Although the surveillance interval is extended by this TS change, this change does not involve a significant reduction in the margin of safety.

The CPC CISAM elements and the various administrative changes included in this TS change do not involve a significant reduction in the margin of safety.

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

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

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

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

*NRC Project Director:* William D. Beckner

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

*Date of amendment request:* December 19, 1996

*Description of amendment request:* Change Request Concerning Addition to the Core Operating Limit Report References

*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:

Criterion 1 - Does not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change to add the technical manual for the Combustion Engineering Nuclear Transient Simulation (CENTS) code to the Core Operating Limits Report (COLR) references is administrative in nature. The CENTS code has been reviewed and approved by the NRC. The physical design or operation of the plant is not impacted by this proposed change. The proposed change does not adversely impact transient analysis assumptions or results. The COLR-related safety analyses will continue to be performed utilizing NRC-approved methodologies, and specific reload changes will be evaluated under the provisions of 10CFR50.59. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change to reference the NRC-approved CENTS code is administrative in nature. No physical alterations of plant configuration, changes to plant operating procedures, or operating parameters are proposed. No new equipment is being introduced, and no equipment is being operated in a manner inconsistent with its design. The COLR-related safety analyses will continue to be performed utilizing NRC-approved methodologies. A 10CFR50.59 safety review will continue to be performed to evaluate specific reload changes. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to reference the CENTS code is administrative in nature. Existing technical specification operability and surveillance requirements are not reduced by the proposed change. The cycle-specific COLR limits for future reloads will continue to be developed based on NRC-approved methodologies. Technical specifications will continue to require that the core be operated within these limits and specify appropriate actions to be taken if the limits are violated. The COLR-related safety analyses will continue to be performed utilizing NRC-approved methodologies, and specific reload changes will be evaluated per 10CFR50.59. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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*NRC Project Director:* William D. Beckner

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

*Date of amendment request:* December 19, 1996

*Description of amendment request:* Change Request Concerning Power Calibration Requirements

*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:

Criterion 1 - Does not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change will redefine the tolerance band allowed for linear power level, the Core Protection Calculator (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, will require more conservative tolerances than are currently allowed. This change will ensure that the power indications are more conservative relative to the existing safety analyses. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change to Technical Specification power calibration tolerance limits are conservative relative to the current requirements. This amendment request does not change the design or operation of any plant systems or components. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The allowed tolerance band for the linear power level, CPC delta T power, and CPC nuclear power signals between 15 and 80% power has been redefined. The new requirements are more conservative than the tolerances that currently exist in the Technical Specifications. This change will ensure that the power indications are more conservative relative to the existing safety analyses. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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*NRC Project Director:* William D. Beckner

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

*Date of amendment request:* December 19, 1996

*Description of amendment request:* Change Request Concerning Reactor Coolant System Volume

*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:

Criterion 1 - Does not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

This proposed change allows the relocation of the reactor coolant system volume in the design features section of technical specifications to the safety analysis report. Future changes will be controlled under 10CFR50.59. This change is considered administrative in nature. Appropriate values of reactor coolant system volume are used in the safety analyses. This change does not affect any system or component functional requirements. The operation of the plant is not affected by this change.

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

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

The relocation of existing requirements from the technical specifications to another licensee controlled document is administrative in nature. This change does not modify or remove any plant design requirement. The proposed change will not affect any plant system or structure, nor will it affect any system functional or operability requirements. Therefore, no new failure modes are introduced as a result of this change.

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

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

The proposed amendment request relocates the coolant system volume located in the technical specifications design feature section to another licensee controlled document, the ANO-2 Safety Analysis Report, which is controlled under 10CFR50.59. The proposed change is administrative in nature because the design requirements for the facility remain the same. The proposed change does not represent a change in the configuration or operation of the plant.

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

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

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

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

*NRC Project Director:* William D. Beckner

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2), Pope County, Arkansas

*Date of amendment request:* December 19, 1996

*Description of amendment request:* Change Control Room Ventillation System Requirements

*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:

Criterion 1 - Does not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The control room emergency ventilation and air conditioning systems are not initiators of an accident previously evaluated. Extension of the allowable outage time for one inoperable control room emergency air conditioning system from 7 days to 30 days is acceptable based on the low probability of an event occurring that would require control room isolation and a concurrent or subsequent failure of the remaining operable control room emergency air conditioning system. An evaluation using probabilistic safety assessment techniques has shown the frequency of this event to be an acceptably low level (4.67E-6/yr). The ANO-1 surveillance requirements for the control room emergency ventilation and air conditioning system have been updated for consistency with the ANO-2 requirements and are consistent with RG 1.52, March 1978, Revision 2 and ASTM D3803-1989. The change in the ANO-2 Mode of Applicability for the control room radiation monitoring instrumentation is acceptable because the only identified accident scenario requiring control room isolation on high radiation while in Modes 5 and 6 is the fuel handling accident and this analysis shows that the dose consequences to the control room operators are acceptable in the event of a fuel handling accident, assuming that the normal control room ventilation system is properly isolated. The remainder of the changes have been made for consistency between the ANO-1 and ANO-2 TS and are considered to be more restrictive or administrative in nature.

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

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

The control room emergency ventilation and air conditioning systems are not accident initiators. The proposed changes introduce no new mode of plant operation and no new possibility for an accident is introduced by modifying the ANO-1 surveillance testing requirements for the control room emergency ventilation and air conditioning systems.

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

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

With the exception of the AOT extension and the relaxation of the ANO-2 Mode of Applicability for the control room radiation monitoring instrumentation, all the ANO-1 and ANO-2 changes are considered administrative or more restrictive and are intended to clarify and make consistent the requirements of the control room emergency habitability equipment. Although the AOT extension does involve an incremental reduction in the margin of safety due to slight increase in the frequency of an event requiring control room isolation, followed by failure of the operable emergency control room chiller, a probabilistic safety assessment has shown this slight increase in frequency (approximately 3.58E-6/yr) to be acceptably low. The change in the ANO-2 Mode of Applicability for the control room radiation monitoring instrumentation is acceptable because the only identified accident scenario requiring control room isolation on high radiation while in Modes 5 and 6 is the fuel handling accident and this analysis shows that the dose consequences to the control room operators are acceptable in the event of a fuel handling accident, assuming that the normal control room ventilation system is properly isolated.

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

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

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

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

*NRC Project Director:* William D. Beckner

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

*Date of amendment request:* December 3, 1996

*Description of amendment request:* The two proposed changes would revise Technical Specification (TS) 2.1.1.2 for Hatch Nuclear Plant, Unit 2, Safety Limit Minimum Critical Power Ratio (SLMCPR) values. The revision is based upon unique plant evaluations for the current Cycle 13 and the use of General Electric (GE) GE-13 fuel, a 9 x 9 fuel design, in the next Cycle 14. The proposed SLMCPRs for Hatch Unit 2 are 1.08 and 1.09 (single-loop operation) for the current Cycle 13, and 1.12 and 1.14 (single-loop operation) for Cycle 14.

The new SLMCPRs were calculated using NRC-approved methods and interim implementing procedures. The SLMCPRs are set high enough to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The SLMCPRs incorporate a margin for uncertainty in the core operating state for uncertainties that are fuel-type dependent, including fuel bundle nuclear characteristics, critical power correlation, and manufacturing tolerances. These interim procedures were revised to incorporate the following cycle-specific parameters: (1) Actual core loading, (2) Conservative variations of projected control blade patterns, (3) Actual bundle parameters (e.g., local peaking), and (4) Full cycle exposure range.

*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 change involve a significant increase in the probability of consequences of an accident previously evaluated?

The derivation of the revised SLMCPRs for Plant Hatch Unit 2 for incorporation into the Technical Specifications, and its use to determine cycle-specific thermal limits, were performed using NRC-approved methods. Additionally, interim implementing procedures incorporating cycle-specific parameters were used. Based upon the use of these calculations, revised SLMCPRs cannot increase the probability or severity of an accident. The basis of the SLMCPR calculation is to ensure that ≤ 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. Thus, it can be concluded that the probability of fuel damage is not increased and the proposed Technical Specifications changes do not involve an increase in the probability or consequences of an accident evaluation.

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

The SLMCPR is a Technical Specifications numerical value designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. The SLMCPRs were calculated using NRC-approved methods. Additionally, interim procedures incorporating cycle-specific parameters were used in the analysis. 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 the margin of safety?

The margin of safety as defined in the Bases will remain the same. The new SLMCPRs were 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 were used. The SLMCPR remains high enough to ensure that ≤ 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. 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:* Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

*NRC Project Director:* Herbert N. Berkow

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

*Date of amendment request:* January 7, 1997

*Description of amendment request:* The proposed amendments would change the Technical Specifications (TS) for Plant Hatch, Units 1 and 2, associated with Surveillance Requirement (SR) testing that requires manually actuating every safety/relief valve (S/RV) during each unit startup from a refueling outage. The proposed changes would provide an alternate method of testing the S/RVs during shutdown conditions rather than during unit startup as is currently done. This approach would reduce valve leakage, thereby reducing the possibility of inadvertent valve actuation and resultant plant transients. Additionally, deletion of testing for the safety mode of the S/RVs is proposed since other testing provides operability verification.

Furthermore, the licensee proposes relief from the applicable requirements of the ASME OM Code (1995), Appendix I, paragraph I 3.4.1(d), which also requires manual actuating of S/RVs during unit startup.

Current Unit 1 and Unit 2 SRs 3.5.1.12 and 3.6.1.6.1 require that each S/RV be manually actuated at pressure conditions. Georgia Power Company (GPC) proposes to revise SRs 3.5.1.12 and 3.6.1.6.1 that would require the S/RVs to be manually actuated in the relief mode during a plant outage before steam is generated. The solenoid valve would be energized, the actuator would stroke, and the pilot rod lift would be measured. This in-situ test would verify that, given a signal to the solenoid, the pilot disc rod would lift. If steam were present, the pilot disc would open and initiate opening of the main stage.

The licensee also proposes to delete current Units 1 and 2 SR 3.4.3.2, which also requires that each S/RV be manually actuated because this test is not necessary to assure S/RV operability in the safety mode since other tests, taken together, confirm the entire S/RV assembly functions adequately.

*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:

Georgia Power Company [GPC] has reviewed the proposed license amendment request and determined its adoption does not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three 10 CFR 50.92 standards follows.

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

Since the proposed Technical Specifications changes and ASME Code relief do not impose any physical changes to the S/RVs, their design function is unaffected. The submittal only proposes changes to the manner in which the S/RVs are tested. As discussed in Enclosure 1 [of the licensee's submittal], the combination of current S/RV testing and the proposed alternate testing will continue to adequately demonstrate the operability of the S/RVs for both the safety and relief modes. Under the proposed testing requirements, it is expected that S/RV leakage will decrease; thus, the probability of occurrence of an inadvertent S/RV actuation is actually reduced.

FSAR [Final Safety Analysis Report] analyzed events, such as MSIV [main steam isolation valve] closure, generator load reject, turbine trip with failure of switchyard breakers to open, and pressure regulator failure, take credit for the S/RVs mitigating the consequences of these events. These proposed changes will not increase the consequences of these events, since a series of S/RV tests (on the bench and installed) will ensure all S/RV components necessary to ensure valve opening will function. The S/RVs will therefore be capable of performing their design functions.

Furthermore, reducing the number of manual actuations of the S/RVs decreases the likelihood of a stuck open S/RV, which is an analyzed event in the Hatch FSAR.

Therefore, the probability of occurrence and the consequences of previously analyzed events are not 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 changes affect the manner in which S/RV operability is verified in that one Technical Specifications SR [surveillance requirement] is being deleted and two are being revised; however, they do not affect the way the S/RVs are operated. The S/RVs will not be operated or tested in a manner contrary to their design. As a result, no new mode of operation is introduced. That is, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The present method of S/RV testing unnecessarily challenges the valves, and is linked to S/RV degradation through pilot valve and/or main valve leakage. This Technical Specifications change should decrease S/RV leakage and improve S/RV reliability by reducing the potential for spurious valve actuation at full power. In this sense, the margin of safety is actually increased; e.g., the likelihood for spurious S/RV actuation is reduced.

Deleting the test of installed S/RVs at rated temperature and pressure will not significantly reduce the margin of safety for events in which S/RV actuation is assumed, since each S/RV receives a series of tests which insure each component necessary for successful opening of the S/RV functions properly. Thus, the S/RV is assured of opening in either the safety or the relief mode. For example, at Wyle Labs, the valves undergo testing at operating steam pressure. This test ensures operability of the pilot and main discs and also verifies set pressure, reseat pressure, and main steam stroke time. As noted previously, upon successful completion of these tests, including verification of zero seat leakage, the valves receive a written certification from the lab and are returned to Plant Hatch for installation.

GPC further proposes that, upon installation, but before steam is generated, the valves receive a test requiring the solenoid to be energized. This test provides additional verification that the pilot disc opens. The remaining segments of the S/RV tests verify the ability of ADS and LLS logic to energize the solenoid.

In summary, this amendment does not involve a significant reduction in the margin of safety, because of the reduction in S/RV degradation, and because remaining tests confirm the valves will function properly when required.

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

*Local Public Document Room location:* Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

*NRC Project Director:* Herbert N. Berkow

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

*Date of amendment request:* December 16, 1996

*Description of amendment request:* The amendment request, if approved, would reflect the change in the legal name of the operator of TMI-1 from GPU Nuclear Corporation to GPU Nuclear Inc. and reflect in the TMI-1 license and the Technical Specifications the registered trade name of GPU Energy.

*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 (SHC), which is presented below:

GPU Nuclear Inc. has determined that the proposed TMI-1 license amendment and technical specification change request involve no significant hazards consideration as defined in 10 CFR 50.92 because:

Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment adds to the license and the technical specifications the trade name of the Owners of TMI-1. The change in the legal name of the operator of TMI-1 is a cosmetic change made to reflect the name changes made throughout the GPU family of companies. The name change has no impact on plant design or operation.

Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated because no new failure modes are created by the proposed changes. The use of a common trade name for the Owners of TMI-1 and the change in the legal name of the operator of TMI-1 has no impact on plant design or operation. Thus, there is no creation of the possibility of a new or different kind of accident from those previously evaluated.

Operation of the facility in accordance with the proposed amendment will not involve a signficant reduction in a margin of safety. The proposed amendment does not change any operating limits for reactor operation.

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:* Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

*Attorney for licensee:* Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037. NRC Acting Project Director: Patrick D. Milano

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook Nuclear Plant, Unit No. 1, Berrien County, Michigan

*Date of amendment request:* August 4, 1995 as supplemented December 20, 1996 [AEP:NRC:1129E and 1129M]

*Description of amendment request:* The proposed amendment would modify the technical specifications (T/S) to allow for repair of hybrid expansion joint (HEJ) sleeves under redefined repair boundary limits. This alternate plugging criterion would assess the integrity of parent tube indications based on the degraded joint geometry, with reference to the specific location of the flaw. The continued operability of the HEJ sleeved tube would be based on the measured diameter difference, or diameter delta (delta D), between the sleeve peak hardroll diameter and the diameter of the sleeve adjacent to the parent tube flaw in the upper joint.

*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:

Conformance of the proposed amendments to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following.

(1) Operation of Cook Nuclear Plant unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The HEJ sleeved tube structural integrity limits defined by this amendment provide for structural integrity consistent with the guidance of RG 1.121. Tube structural integrity consistent with the most limiting RG 1.121 loading is inherently provided by a measured [delta D] of less than 1 mil, although the criterion specifies a minimum of 3 mils must be verified. The structural integrity characteristics of a postulated degraded parent tube with a 3 mil [delta D] provides for axial restraint capability of more than double the most limiting RG 1.121 loading, which indicates that the postulated separated tube would not become axially displaced relative to the sleeve during any plant condition.

Based on tube pull data from Cook Nuclear Plant and other plants it is expected that TSP intersections would provide a substantial axial restraint capability. This interaction is neglected in the analysis of the criterion, and provides for extra safety margin.

Based on the destructive examination results for sections of HEJ sleeved tubes removed in 1994 from another plant, the parent tube flaw morphology is described as circumferentially oriented with multiple initiation sites. This segmented morphology indicates that the previously performed structural capability testing is conservative. Additional axial load bearing capability is provided by the segmented morphology since end cap loading would be transmitted through the tube by the non-degraded ligaments of the segmented crack network, and tube separation therefore, is not likely or credible.

The consequences of any postulated failure of a sleeved tube to which the criteria has been applied would be bounded by the current steam generator tube rupture event discussed in the Cook Nuclear Plant Final Safety Analysis Report (FSAR). Axial displacement of any tube, sleeved or unsleeved, is bounded by approximately 1.1 inch. A tube which experiences axial displacement by this amount would be expected to exhibit a release rate well below the normal makeup capacity. In order for a HEJ sleeved tube to exhibit reactor coolant system release rates approaching the release rates assumed in the FSAR the tube must be displaced by approximately 3 inches. In order for the postulated separated tube to experience axial displacement of any magnitude, it must be assumed that the HEJ hardroll provides no structural benefit and that the tube-to-TSP interaction is frictionless.

Postulated primary to secondary leakage during a main steam line break event will be assessed against the limit of 8.4 gpm in the faulted loop, calculated as part of the voltage based plugging limit for tube support plate intersections. The total of all leakage sources must be shown to be less than this value.

Application of the 3 mil [delta D] criterion (excluding eddy current uncertainty) does not change existing reactor coolant system flow conditions, therefore, existing LOCA analysis results will be unaffected. Plant response to design basis accidents for the current tube plugging and flow conditions are not affected by the repair process; no new tube diameter restriction is introduced.

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

Application of the proposed 3 mil [delta D] HEJ sleeved tube structural integrity criterion will not introduce significant or adverse changes to the plant design basis. The 3 mil [delta D] criteria provides for structural integrity of the HEJ sleeved tube assembly which significantly exceeds the limiting RG 1.121 loading condition. Under these conditions neither a single nor a multiple tube rupture event is considered credible.

The general outline of the HEJ sleeve is unaffected, and the application of the proposed criterion does not change the sleeve configuration or size/shape. The application of the criterion also does not represent a potential to affect other plant components.

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

The proposed criterion has been shown to provide structural integrity of the tube bundle consistent with the most limiting RG 1.121 tube integrity recommendations. In order for tube rupture to occur, the degraded parent tube must experience a complete circumferential separation and be subsequently axially displaced by approximately 3 inches. The inherent structural integrity provided by the interference fit of the HEJ in addition to the axial restraint provided by tube support plate intersections above the HEJ provides for structural integrity far exceeding the RG 1.121 loading of 2264 lb. Even in the event that a degraded HEJ sleeved parent tube were to experience axial displacement, the maximum amount of displacement the tube could experience is bounded by 1.11 inch. Postulating that the tube were to become displaced by this amount, primary to secondary leakage would be limited to well less than the normal makeup capacity due to the proximity between the hydraulically expanded sleeve OD and tube ID.

Pulled HEJ sleeved tube samples from another plant with HEJ sleeved tubes indicate that the crack morphology is described as circumferentially oriented cracking with multiple initiation sites. This segmented morphology provides for additional structural margin not modeled in the testing program.

Existing flow equivalency calculations for the HEJ sleeved tubes will be unaffected by the application of the criterion.

Based on the preceding analysis it is concluded that operation of Cook Nuclear Plant unit 1 following the application of the 3 mil [delta D] HEJ sleeved tube structural integrity limit does not increase the probability of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or reduce any margins to plant safety. Therefore, the license amendment does not involve a significant hazards consideration as defined in 10 CFR 50.92.

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

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

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

*NRC Project Director:* Gail H. Marcus

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

*Date of application for amendment:* October 30, 1996

*Description of amendment request:* These amendments revise the safety limit minimum critical power ratios (SLMCPRs) at Peach Bottom Atomic Power Station, Unit 3.

*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 TS [technical specification] changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The derivation of the cycle-specific SLMCPRs for incorporation into the TS, and its use to determine cycle-specific thermal limits, have been performed using USNRC [U.S. Nuclear Regulatory Commission]-approved methods as discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE-24011-P-A-11, and U.S. Supplement, NEDE-24011-P-A-11-US, November 17, 1995 and interim (reconfirmation) implementing procedures. This change in SLMCPRs cannot increase the probability or severity of an accident.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. The fuel licensing acceptance criteria for the SLMCPR calculation apply to PBAPS [Peach Bottom Atomic Power Station], Unit 3, Cycle 11 in the same manner as they have applied previously. The probability of fuel damage is not increased. Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR is a TS numerical value, designed to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core during the limiting postulated accident. It cannot create the possibility of any new type of accident. The new SLMCPRs are calculated using USNRC-approved methods (≥General Electric Standard Application for Reactor Fuel,'' NEDE-24011-P-A-11, and U.S. Supplement, NEDE-24011-P-A-11-US, November 17, 1995) and interim (reconfirmation) implementing procedures.

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 SLMCPRs are calculated using USNRC-approved methods (≥General Electric Standard Application for Reactor Fuel,'' NEDE-24011-P-A-11, and U.S. Supplement, NEDE-24011-P-A-11-US, November 17, 1995) and interim (reconfirmation) implementing procedures which are in accordance with the current fuel licensing criteria. The SLMCPRs 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. Therefore, the proposed TS changes do not involve a reduction in a margin of safety.

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

*Local Public Document Room location:* Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

*NRC Project Director:* John F. Stolz

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

*Date of amendment request:* January 15, 1997

*Description of amendment request:* The amendment proposes to relocate the snubber operability, surveillance, and record requirements for components (snubbers) in the Technical Specifications (TS) to plant controlled documents.

*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:

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, based on the following:

1. These changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

The changes relocate operability, surveillance, and record requirements for components (snubbers) which do not meet the criteria for inclusion in the Technical Specifications (TS). The affected components are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The snubber requirements will be relocated from the TS to plant controlled documents. These requirements will be maintained pursuant to 10 CFR 50.59. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The changes do not create the possibility of a new or different type of accident previously evaluated because:

The changes do not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or affect parameters governing normal plant operation. Adequate control of future changes to snubber requirements will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated for the plant.

3. The proposed changes do not involve a reduction in a margin of safety because:

The changes do not involve a change to the operability, surveillance, and record requirements for the snubber program as they currently exist in the TS, nor do they impact on any safety analysis assumptions. The proposed changes relocate snubber requirements from the TS to plant controlled documents. Changes to the requirements in these documents are subject to the requirements of 10 CFR 50.59. In addition, exceptions to code requirements for testing will require NRC approval. Regulations and

FitzPatrick commitments to the NRC contain the necessary programmatic requirements for the plant controlled documents. Operating limitations will continue to be imposed, and required surveillances will continue to be performed in accordance with regulations, FitzPatrick commitments to the NRC, and written procedures and instructions that are auditable by the NRC. If snubber inoperability causes a TS system or component to be inoperable, then the affected system or component Limiting Condition for Operation (LCO) will be entered. Based on the above, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

*Attorney for licensee:* Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

*NRC Project Director:* S. Singh Bajwa, Acting Director

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 amendment request:* January 7, 1997

*Description of amendment request:* The proposed amendment would revise Technical Specification (TS) 3/4.2.5 to incorporate an exception to the provisions of TS 4.0.4 and to clarify the time at which the surveillance can be performed by adding that the surveillance is to be performed within 24 hours after attaining steady state conditions at or above 90% rated thermal power. The revised surveillance would also contain editorial enhancements that do not change the intent of the current surveillance. TS Table 3.2-1 for Salem Unit 1 would be revised to delete reference to three loop operation (which is not permitted at Salem Unit 1) in order to eliminate potential confusion when applying this table.

*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 changes proposed on the RCS [Reactor Coolant System] flow measurement and exemption to Specification 4.0.4 do not affect the operation of the equipment during conditions when they are required to perform their safety function. No physical changes to the plant result from the proposed changes made to the surveillance requirements. The measurement of RCS flow does not impact the probability of an accident.

Testing is being performed with the plant in the condition in which the automatic initiation signals for low RCS flow would result in a time consistent with the TS requirements.

Protection System in providing a reactor trip upon a loss of RCS flow. Degradations in flow will occur over a long duration; however, testing will continue to be performed within twenty-four hours upon achieving steady state greater than or equal to 90% RTP [Rated Thermal Power] after refueling which is a sufficiently short duration after startup to identify flow degradations.

Changes proposed to refer to Table 3.2-1 for the DNB [Departure from Nucleate Boiling] parameters and to delete the Unit 1 three loop operation parameters, and the inclusion of the type of test performed are editorial in nature.

Therefore, the consequences of an accident previously evaluated are not significantly increased by the proposed changes.

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

The proposed changes do not involve any modifications to existing plant equipment, do not alter the function of any plant systems, do not introduce any new operating configurations or new modes of plant operation, nor change the safety analyses. The point at which RCS flow is measured using a heat balance will not impact the ability to maintain or monitor Reactor Coolant flows. The proposed changes will, therefore, not create the possibility of a new or different kind of accident from any accident previously evaluated.

Changes proposed to refer to Table 3.2-1 for the DNB parameters and to delete the Unit 1 three loop operation parameters, and the inclusion of the type of test performed are editorial in nature.

[The proposed changes will, therefore, 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 changes to the RCS flow surveillance do not decrease the scope of the existing testing, but will clarify the point at which the testing is performed.

The time in which testing is performed, after achieving steady state conditions after reaching greater than or equal to 90% RTP ensures that testing is performed in a timely manner. Flow margins established as a result of previous testing will not be significantly reduced in light of recent outage activities. Future changes that might impact margins established by the testing will be reviewed in accordance with the requirements of 10 CFR 50.59.

Changes proposed to refer to Table 3.2-1 for the DNB parameters and to delete the Unit 1 three loop operation parameters, and the inclusion of the type of test performed are editorial in nature.

All changes are consistent with the intent of Salem's current TS [Technical Specification] and with the 18 month surveillances specified in NUREG-1431, Revision 1.

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

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

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

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

*NRC Project Director:* John F. Stolz

Southern Nuclear Operating Company, Inc., Docket No. 50-348, Joseph M. Farley Nuclear Plant, Unit 1, Houston County, Alabama

*Date of amendments request:* December 26, 1996

*Description of amendments request:* The proposed amendment would revise Technical Specification 3/4.4.6 ''Steam Generators'' and its associated Bases. Specifically, the steam generator repair limit would be modified to clarify that the appropriate method for determining serviceability for tubes with outside diameter stress corrosion cracking at the tube support plate is by a methodology that more reliably assesses structural integrity. This amendment request is in accordance with NRC's Generic Letter 95-05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.''

*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) Operation of Farley units in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Testing of model boiler specimens for free standing tubes at room temperature conditions shows burst pressures as high as approximately 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 26.5 volts. Burst testing performed on pulled tubes, including tubes pulled from Farley Unit 1, with up to 7.5 volt indications show burst pressures in excess of 5300 psi at room temperature. ... [T]ube burst criteria are inherently satisfied during normal operating conditions by the presence of the tube support plate. Furthermore, correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the R.G. [Regulatory Guide] 1.121 criterion requiring the maintenance of a margin of 1.43 times the steam line break pressure differential on tube burst if through-wall cracks are present without regard to the presence of the tube support plate. Considering the existing data base, this criterion is satisfied with bobbin coil indications with signal amplitudes over twice the 2.0 volt voltage-based repair criteria, regardless of the indicated depth measurement. This structural limit is based on a lower 95% confidence level limit of the data at operating temperatures. The 2.0 volt criterion provides an extremely conservative margin of safety to the structural limit considering expected growth rates of outside diameter stress corrosion cracking at Farley. Alternate crack morphologies can correspond to a voltage so that a unique crack length is not defined by a burst pressure to voltage correlation. However, relative to expected leakage during normal operating conditions, no field leakage has been reported from tubes with indications with a voltage level of under 7.7 volts for a 3/4 inch tube with a 10 volt correlation to 7/8 inch tubing (as compared to the 2.0 volt proposed voltage-based tube repair limit). Thus, the proposed amendment does not involve a significant increase in the probability or consequences of an accident.

Relative to the expected leakage during accident condition loadings, the accidents that are affected by primary-to-secondary leakage and steam release to the environment are Loss of External Electrical Load and/or Turbine Trip, Loss of All AC Power to Station Auxiliaries, Major Secondary System Pipe Failure, Steam Generator Tube Rupture, Reactor Coolant Pump Locked Rotor, and Rupture of a Control Rod Drive Mechanism Housing. Of these, the Major Secondary System Pipe Failure is the most limiting for Farley in considering the potential for off-site doses. The offsite dose analyses for the other events which model primary-to-secondary leakage and steam releases from the secondary side to the environment assume that the secondary side remains intact. The steam generator tubes are not subjected to a sustained increase in differential pressure, as is the case following a steam line break event. This increase in differential pressure is responsible for the postulated increase in leakage and associated offsite doses following a steam line break event. In addition, the steam line break event results in a bypass of containment for steam generator leakage. Upon implementation of the voltage-based repair criteria, it must be verified that the expected distributions of cracking indications at the tube support plate intersections are such that primary-to-secondary leakage would result in site boundary dose within the current licensing basis. Data indicate that a threshold voltage of 2.8 volts could result in through-wall cracks long enough to leak at steam line break conditions. Application of the proposed repair criteria requires that the current distribution of a number of indications versus voltage be obtained during the refueling outages. The current voltage is then combined with the rate of change in voltage measurement and a voltage measurement uncertainty to establish an end of cycle voltage distribution and, thus, leak rate during steam line break pressure differential. The leak rate during a steam line break is further increased by a factor related to the probability of detection of the flaws. If it is found that the potential steam line break leakage for degraded intersections planned to be left in service coupled with the reduced allowable specific activity levels result in radiological consequences outside the current licensing basis, then additional tubes will be plugged or repaired to reduce steam line break leakage potential to within the acceptance limit. Thus, the consequences of the most limiting design basis accident are constrained to present licensing basis limits, and therefore there is no change to the probability or consequences of an accident previously evaluated.

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

Implementation of the proposed voltage-based tube repair criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism that could result in an accident outside of the region of the tube support plate elevations. Neither a single or multiple tube rupture event would be expected in a steam generator in which the repair criteria have been applied during all plant conditions. The bobbin probe signal amplitude repair criteria are established such that operational leakage or excessive leakage during a postulated steam line break condition is not anticipated. Southern Nuclear has previously implemented a maximum leakage limit of 140 gpd per steam generator. The R.G. 1.121 criterion for establishing operational leakage limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture. The 140 gpd limit provides for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. R.G. 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at steam line break pressure differential. A voltage amplitude of approximately 9 volts for typical outside diameter stress corrosion cracking corresponds to meeting this tube burst requirement at the 95% prediction interval on the burst correlation. Alternate crack morphologies can correspond to a voltage so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, a typical burst pressure versus through-wall crack length correlation is used below to define the ‘‘longest permissible crack’’ for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at 1.43 times steam line break pressure differential and steam line break conditions are about 0.54 inch and 0.84 inch, respectively. Normal leakage for these crack lengths would range from about 0.4 gallons per minute to 4.5 gallons per minute, respectively, while lower 95% confidence level leak rates would range from about 0.06 gallons per minute to 0.6 gallons per minute, respectively.

An operating leak rate of 140 gpd per steam generator has been implemented. This leakage limit provides for detection of 0.4 inch long cracks at nominal leak rates and 0.6 inch long cracks at the lower 95% confidence level leak rates. Thus, the 140 gpd limit provides for plant shutdown prior to reaching critical crack lengths for steam line break conditions at leak rates less than a lower 95% confidence level and for three times normal operating pressure differential at less than nominal leak rates.

Considering the above, the implementation of voltage-based repair criteria will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The use of the voltage-based repair criteria is demonstrated to maintain steam generator tube integrity commensurate with the requirements of Generic Letter 95-05 and R.G. 1.121. R.G. 1.121 describes a method acceptable to the NRC staff for meeting GDC [General Design Criteria] 2, 14, 15, 31, and 32 by reducing the probability of the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of outside diameter stress corrosion cracking at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The most limiting effect would be a possible increase in leakage during a steam line break event. Excessive leakage during a steam line break event, however, is precluded by verifying that, once the criteria are applied, the expected end of cycle distribution of crack indications at the tube support plate elevations would result in minimal, and acceptable primary to secondary leakage during the event and, hence, help to demonstrate radiological conditions are less than an appropriate fraction of the 10 CFR [Part] 100 guideline.

The margin to burst for the tubes using the voltage-based repair criteria is comparable to that currently provided by existing technical specifications.

In addressing the combined effects of LOCA [loss-of-coolant accident] + SSE [safe shutdown earthquake] on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to either the LOCA rarefaction wave and/or SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential the partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse or that short through-wall indications would leak at significantly higher leak rates than included in the leak rate assessments.

Consequently, a detailed leak-before-break analysis was performed and it was concluded that the leak-before-break methodology (as permitted by GDC 4) is applicable to the Farley reactor coolant system primary loops and, thus, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were analyzed at Farley and were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation.

Regardless of whether or not leak-before-break is applied to the primary loop piping at Farley, any flow area reduction is expected to be minimal (much less than 1%) and PCT margin is available to account for this potential effect. Based on analyses' results, no tubes near wedge locations are expected to collapse or deform to the degree that secondary to primary in-leakage would be increased over current expected levels. For all other steam generator tubes, the possibility of secondary-to-primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary-to-primary leakage in the event of a LOCA + SSE is expected to be less than that originally allowed, i.e., 500 gpd per steam generator. Furthermore, secondary-to-primary in-leakage would be less than primary-to-secondary leakage for the same pressure differential since the cracks would tend to tighten under a secondary-to-primary pressure differential. Also, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing the R.G. 1.83 considerations, implementation of the tube repair criteria is supplemented by 100% inspection requirements at the tube support plate elevations having outside diameter stress corrosion cracking indications, reduced operating leakage limits, eddy current inspection guidelines to provide consistency in voltage normalization, and rotating probe inspection requirements for the larger indications left in service to characterize the principle degradation mechanism as outside diameter stress corrosion cracking.

As noted previously, implementation of the voltage-based repair criteria will decrease the number of tubes that must be taken out of service with tube plugs or repaired. The installation of steam generator tube plugs or tube sleeves would reduce the RCS flow margin, thus implementation of the voltage-based repair criteria will maintain the margin of flow that would otherwise be reduced through increased tube plugging or sleeving.

Considering the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any bases of the plant Technical Specifications.

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

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

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

*NRC Project Director:* Herbert N. Berkow

Southern Nuclear Operating Company, Inc., Docket No. 50-348, Joseph M. Farley Nuclear Plant, Unit 1, Houston County, Alabama

*Date of amendments request:* January 10, 1997

*Description of amendments request:* The proposed amendments would implement repair of tubes using laser welded tube sleeves for the steam generators at Farley Units 1 and 2 as described in WCAP-13088, Revision 4, and WCAP-14740. In addition, for Unit 2, references to a one-cycle limited implementation of L* are being removed. The approval for the limited implementation of L* expired at the last Unit 2 outage in the fall of 1996.

*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. Operation of Farley Units 1 and 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The laser welded sleeve configurations as described within WCAP-13088, Revision 4 and WCAP-14740 have been designed and analyzed in accordance with the requirements of the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code]. Fatigue and stress analyses of the sleeved tube assemblies produced acceptable results. Mechanical testing has shown that the structural strength of Alloy 690 sleeves under normal, faulted and upset conditions is within acceptable limits. Leakage testing for 7/8 inch tube sleeves has demonstrated that significant primary-to-secondary leakage is notexpected during all plant conditions, including the case where the seal weld is not produced in the lower joint of the tubesheet sleeve.

Initial acceptance of welded joints uses ultrasonic inspection to verify that all weld thicknesses meet the minimum specified conditions over the entire circumference. A plugging limit of 24% allowable depth of penetration of the sleeve tube wall thickness applies for each type of laser welded sleeve that may be installed in the Farley Nuclear Plant steam generators and is determined for uprated conditions with a limiting steam pressure for reduced $T_{hot}$ and 20% steam generator tube plugging conditions. These conditions represent the limiting primary-to-secondary operating pressure differential, which is bounding for the sleeve plugging limit and structural analysis inputs. However, the state-of-the-art in eddy current inspection capability is such that no probes are qualified to size the depth of penetration of stress corrosion cracking. It is generally believed that the detection threshold of these probes is well below 40% throughwall. Southern Nuclear Operating Company will plug on detection any crack-like indications that may occur in the sleeve using the sleeve inspection probe of record until an inspection process is qualified to size depth of penetration of stress corrosion cracking into the tube wall.

The hypothetical consequences of failure of the sleeve would be bounded by the current steam generator tube rupture analysis included in the Farley Nuclear Plant FSAR [Final Safety Analysis Report]. Due to the slight reduction in diameter caused by the sleeve wall thickness, it is expected that primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system. Combinations of tubesheet sleeves and tube support plate sleeves would reduce the primary fluid flow through the sleeved tube assembly due to the series of diameter reductions the fluid would have to pass on its way to the break area. The overall effect would be reduced steam generator tube rupture release rates.

As addressed previously, the proposed Technical Specification change to support the installation of full length tubesheet, elevated tubesheet, or tube support plate elevation Alloy 690 laser welded sleeves as described in WCAP-13088, Revision 4 and WCAP-14740 does not adversely impact any other previously evaluated design basis accident or the results of LOCA [loss-of-coolant accident] and non-LOCA accident analyses for the current Technical Specification minimum reactor coolant system flow rate. The results of the analyses and testing, as well as plant operating experience, demonstrate that the sleeve assembly is an acceptable means of restoring tube integrity to a condition consistent with its original design basis. Also, per Regulatory Guide 1.83, Revision 1 recommendations, the condition of the sleeved tube can be monitored through periodic inspections with present eddy current techniques.

Conformance of the sleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests support the conclusion that the installation of laser welded tube sleeves will not increase the probability or consequences of an accident previously evaluated. Depending upon the break location for a postulated steam generator tube rupture event, implementation of tube sleeving could act to reduce the radiological consequences to the public due to reduced primary to secondary flow rate through a sleeved tube compared to a non-sleeved tube based on the restriction afforded by the sleeve wall thickness.

Removal of the references to the interim use of an L* repair criteria will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Implementation of laser welded sleeving will not introduce significant or adverse changes to the plant design basis. Sleeving also does not represent a potential to affect any other plant component. Stress and fatigue analysis of the repair has shown the ASME Code minimum stress values are not exceeded. Implementation of laser welded sleeving maintains overall tube bundle structural and leakage integrity at a level consistent to that of the originally supplied tubing during all plant conditions. Leak and mechanical testing of sleeves support the conclusions of the calculations that each sleeve joint retains both structural and leakage integrity during all conditions. Sleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the sleeves. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. Since the sleeve design does not affect any other component or location of the tube outside of the immediate area repaired, in addition to the fact that the installation of sleeves and the impact on current plugging level analyses is accounted for, the possibility that laser welded sleeving creates a new or different type of accident is not credible.

Removal of the references to the interim use of an L* repair criteria will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The laser welded sleeving repair of degraded steam generator tubes as identified in WCAP-13088, Revision 4, has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition as the requirements of the ASME Code are satisfied. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The design of the tubesheet sleeve lower joints for the 7/8 inch sleeves (for both the full length and elevated tubesheet sleeve) have been verified by testing to preclude realistic leakage during normal and postulated accident conditions.

The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the recommendations of Regulatory Guide 1.83, Revision 1 and the surveillance requirements included in Specification 4.4.6.0. The portion of the tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The areas of the sleeved tube assembly which require inspection are defined in WCAP-13088, Revision 4.

The effect of sleeving on the design transients and accident analyses have been reviewed based on the installation of sleeves up to the level of steam generator tube plugging coincident with the minimum reactor flow rate. The installation of sleeves is to be evaluated as the equivalent of some level of steam generator tube plugging. Evaluation of the installation of sleeves is based on the determination that LOCA evaluations for the licensed minimum reactor coolant flow bound the effect of a combination of tube plugging and sleeving up to an equivalent of the actual steam generator tube plugging limit. Information provided in WCAP-13088, Revision 4, describes the method to determine the flow equivalency for all combinations of tubesheet and tube support plate sleeves in order that the minimum flow requirements are met.

Implementation of laser welded sleeving will reduce the potential for primary-to-secondary leakage during a postulated steam line break while maintaining available primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube through repair, primary pressure boundary integrity is restored and the potential for primary-to-secondary leakage during all plant conditions is minimized. These degraded tubes are returned to a condition consistent with the design basis. While the installation of a sleeve causes a reduction in primary coolant flow, the reduction is significantly below the reduction incurred by plugging. Therefore, greater primary coolant flow area is maintained through sleeving.

Removal of the references to the interim use of an L* repair criteria will 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:* Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

*NRC Project Director:* Herbert N. Berkow

Tennessee Valley Authority, Docket No. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

*Date of amendment request:* January 10, 1997

*Description of amendment request:* The proposed amendment would modify the Watts Bar Nuclear Plant (WBN) Unit 1 Technical Specifications (TS) in order to implement the 1995 rule change to 10 CFR Part 50, Appendix J. The revised Appendix J provided an Option B which allows performance based testing for containment leakage rate testing. The TS in Section 3.6 and associated Bases, TS Section 3.0.2 and TS Section 5.7 would be changed. Also, the schedular exemption for containment airlock testing now specified in the facility license in Section 2.D(1) would no longer be required and would be deleted.

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

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

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

The proposed amendment to WBN TSs is in accordance with the new performance-based option (Option B) to 10 CFR 50,

Appendix J. The changes being proposed will not change the physical plant or the modes of operation defined in the facility license. The proposed changes do not increase the total allowable primary containment leakage rate. The changes do not involve the addition or modification of equipment, nor do they alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the propsoed change does not create the possibility of a new or diferent kind of accident from any previously evaluated.

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

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

The NRC 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:* Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402

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

*NRC Project Director:* Frederick J. Hebdon

## 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.

### 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:* November 26, 1996, as supplemented December 17, 1996

*Description of amendment request:* The proposed amendments would allow a one-time only change necessary to replace the existing 125-volt dc battery cells with new cells. Date of publication of individual notice in Federal Register: December 13, 1996 (61 FR 65605)

*Expiration date of individual notice:* January 13, 1997

*Local Public Document Room location:* J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina 28223-0001

## 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.

### Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

*Date of application for amendment:* August 23, 1996

*Brief description of amendment:* This amendment makes Technical Specifications changes allowing fuel enrichment of up to 5.0 weight percent Uranium-235. The previous limit was 4.1 weight percent. This change allows Arkansas Nuclear One, Unit-2, to receive, store, and use nuclear fuel of 5.0 weight percent Uraninum-235.

*Date of issuance:* January 14, 1997

*Effective date:* January 14, 1997

*Amendment No.:* 178

*Facility Operating License No.* NPF-6. Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: October 9, 1996 (61 FR 52964) 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:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

### Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

*Date of application for amendment:* November 24, 1996, as supplemented on December 2, 1996.

*Brief description of amendment:* This amendment adds small break-loss-of coolant accident methodology CENPD-137, Supplement 1-P and its approval letter by the NRC as a reference to Section 6.9.5.1. This code previously approved by the NRC increases the steam generator tube plugging limit to 30% with an associated reduction of 10% in RCS flow. This amendment also corrects a typographical error in Specification 6.9.5.1.8, and Specifications 6.9.5.1.10 through 6.9.5.1.14 are numbered to accommodate these changes.

*Date of issuance:* January 14, 1997

*Effective date:* January 14, 1997

*Amendment No.:* 179

*Facility Operating License No.* NPF-6. Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: December 3, 1996 (61 FR 64173) However, on December 9, 1996, the licensee verified that the number of plugged tubes would not exceed their current 10% limit established by the old code. This determination removed the basis for considering this request as exigent. Since the potential does exist for the plugging to exceed the 10% in the future, the technical specification amendment request is therefore, a valid request on a normal schedule. This change did not alter the staff's initial proposed no safety hazard condition determination, therefore noticing was not warranted. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 14, 1997.

*Local Public Document Room location:* Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

*Date of amendment request:* October 24, 1996

*Brief description of amendment:* The amendment revises the technical specifications to delete the accelerated testing requirements for the standby diesel generators. This action is consistent with the provisions of Generic Letter 94-01, ''Removal of Accelerated Testing and Special Reporting Requiremets for Emergency Diesel Generators,'' dated May 31, 1994.

*Date of issuance:* January 14, 1997

*Effective date:* January 14, 1997

*Amendment No.:* 90

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

*Date of initial notice in* Federal Register: December 4, 1996 (61 FR 64384) 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:* Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

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

*Date of amendment request:* May 30, 1996

*Brief description of amendment:* The amendment revises the technical specification surveillance requirement 3.8.3.4 to specify a 5-start pressure for the air recievers associated with the Division III, High Pressure Core Spray emergency diesel generator.

*Date of issuance:* January 16, 1997

*Effective date:* January 16, 1997

*Amendment No.:* 91

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

*Date of initial notice in* Federal Register: July 3, 1996 (61 FR 34892) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 1997.No significant hazards consideration comments received. No.

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

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

*Date of application for amendment:* September 23, 1996

*Brief description of amendment:* Changes to Technical Specification (TS) to delete a note for the Surveillance Requirement 3.3.7.1 for the Engineered Safeguard Actuation System Logic.Date of issuance: January 6, 1997

*Effective date:* January 6, 1997

*Amendment No.:* 155

*Facility Operating License No.* DPR-72. Amendment revised the TS.

*Date of initial notice in* Federal Register: October 23, 1966 (61 FR 55034) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 6, 1997.No significant hazards consideration comments received: No.

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

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

*Date of amendment request:* October 23, 1996, as supplemented by letter dated November 6, 1996.

*Brief description of amendments:* The amendments revised Technical Specification 3.4.6.1, regarding reactor coolant system leakage detection instrumentation, to adopt the requirements found in NUREG-1431, ''Standard Technical Specifications Westinghouse Plants,'' for the reactor coolant system leakage detection instrumentation.

*Date of issuance:* January 8, 1997

*Effective date:* January 8, 1997

*Amendment Nos.:* Unit 1 - Amendment No. 86; Unit 2 - Amendment No. 73

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

*Date of initial notice in* Federal Register: December 4, 1996 (61 FR 64387) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 8, 1997.No significant hazards consideration comments received: No

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

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:* April 19, 1996, and as supplemented on August 15, 1996

*Brief description of amendment:* The amendment introduces new Technical Specification (TS) 3.10.10, ''Single Control Rod Withdrawal - Refueling,'' under TS 3.10, ''SPECIAL OPERATIONS.'' The purpose of this Special Operations LCO is to permit the withdrawal of a single control rod for testing in MODE 5 without imposing the requirements for establishing the secondary containment and main control room boundaries as normally required during Core Alterations.

*Date of issuance:* January 13, 1997

*Effective date:* January 13, 1997

*Amendment No.:* 112

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

*Date of initial notice in* Federal Register: May 22, 1996 (61 FR 25707) and September 25, 1996 (61 FR 50344). The August 15, 1996, submittal changed the focus of the original amendment request, therefore, it was re-noticed in the FEDERAL REGISTER. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 13, 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, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

*Date of application for amendment:* September 12, 1995

*Brief description of amendment:* The amendment revises Technical Specification 6.3.1 to add a requirement that the Assistant Operations Manager hold a senior reactor operator (SRO) license if the Operations Manager does not hold an SRO license for Millstone Unit 3.

*Date of issuance:* January 7, 1997

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

*Amendment No.:* 132

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

*Date of initial notice in* Federal Register: March 27, 1996 (61 FR 13530) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 7, 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

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:* April 26, 1996, as supplemented August 23, 1996.

*Brief description of amendment:* The amendment changes requirements regarding reactor coolant system leakage testing following refueling outage and other sytem pressure testing of reactor coolant system following repairs, replacements, or modifications.

*Date of issuance:* January 7, 1997

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

*Amendment No.:* 171

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

*Date of initial notice in* Federal Register: June 5, 1996 (61 FR 28602) The August 23, 1996, letter provided clarifying information that did not change the initial no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 7, 1997No significant hazards consideration comments received: No

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

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

*Date of application for amendment:* October 29, 1996

*Brief description of amendment:* The amendment corrects an error with respect to Table 3.3.2-1, Function 6c of the Technical Specifications (TSs) which references the incorrect Required Action for inoperable channels of the auxiliary feedwater pump actuation on Steam Generator Level - Low Low logic. The TSs are revised to correct the Required Action to place the inoperable channel in ''trip'' within 6 hours or initiate a plant shutdown to Mode 4.

*Date of issuance:* January 9, 1997

*Effective date:* January 9, 1997

*Amendment No.:* 66

*Facility Operating License No.* DPR-18: Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: December 4, 1996 (61 FR 64395) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 1997No significant hazards consideration comments received: No

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

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

*Date of application for amendment:* October 29, 1996

*Brief description of amendment:* This amendment revises the MODE of applicability for the motor-driven auxiliary feedwater pump actuation on opening of the main feedwater pump breakers to correct an error introduced during Amendment No. 61.

*Date of issuance:* January 9, 1997

*Effective date:* January 9, 1997

*Amendment No.:* 67

*Facility Operating License No.* DPR-18: Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: December 4, 1996 (61 FR 64395) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 1997No significant hazards consideration comments received: No

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

*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:* April 29, 1996, as supplemented October 21, December 2, and December 16, 1996

*Brief description of amendments:* These amendments revise Technical Specification (TS) Section 15.3.14, ''Fire Protection System,'' and Section 15.4.15, ''Fire Protection System,'' and relocate the requirements of the fire protection program from the TS and incorporate, by reference, the NRC-approved fire protection program into the Final Safety Analysis Report. In addition, the amendments revise the operating licenses to include the NRC's standard fire protection condition. The amendments also approve administrative changes consistent with the relocation as well as corrections to several typographical errors.

*Date of issuance:* January 8, 1997

*Effective date:* January 8, 1997, and implementation within 90 days from the date of issuance. Implementation shall include the relocation of Technical Specification requirements to the appropriate licensee-controlled document as identified in the licensee's application dated April 29, 1996, as supplemented October 21, December 2, and December 16, 1996, and reviewed in the staff's safety evaluation dated January 8, 1997.

*Amendment Nos.:* Unit 1 - 170, Unit 2 - 174

*Facility Operating License Nos.* DPR-24 and DPR-27: Amendments revise the Technical Specifications and the operating licenses.

*Date of initial notice in* Federal Register: June 5, 1996 (61 FR 28621) The October 21, December 2, and December 16, 1996, supplements provided corrected license and TS pages and a 90-day implementation schedule. These supplements were within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 8, 1997.No significant hazards consideration comments received: No

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

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

*Date of application for amendment:* October 31, 1996

*Brief description of amendment:* The amendment revises Kewaunee Nuclear Power Plant Technical Specification 6.9, ''Reporting Requirements,'' by deleting the annual requirement to submit a description of changes made pursuant to 10 CFR 50.59. Administrative changes are also made to correct inconsistencies in the TS Table of Contents and in a footnote for Table TS 3.5-1.

*Date of issuance:* January 6, 1997

*Effective date:* January 6, 1997

*Amendment No.:* 131

*Facility Operating License No.* DPR-43: Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: December 4, 1996 (61 FR 64397) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 6, 1997.No significant hazards consideration comments received: No.

*Local Public Document Room location:* University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Dated at Rockville, Maryland, this 22nd day of January 1997.

For The Nuclear Regulatory Commission

Elinor G. Adensam,

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

[Doc. 97–1994 Filed 1-28-97; 8:45 am]

**BILLING CODE 7590-01-F**

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## [Docket Nos. 50–255 and 72–7]

## Consumers Power Co., Palisades Nuclear Plant, License Nos. DPR–20; Issuance of Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Acting Director, Office of Nuclear Reactor Regulation, has issued a Director's Decision concerning a Petition dated September 19, 1995, as amended on September 30, 1996, filed by Don't Waste Michigan and Lake Michigan Federation (Petitioners) under Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206). The Petition requested that the NRC (1) find that Consumers Power Company (licensee) violated NRC requirements related to unloading procedures for dry storage casks for spent nuclear fuel, (2) suspend the licensee's use of the general license provisions related to dry cask storage of spent nuclear fuel, (3) require a substantial penalty be paid by the licensee, and (4) conduct hearings related to unloading procedures for dry storage casks at Palisades.

The Acting Director of the Office of Nuclear Reactor Regulation has determined that Petition should be granted in part and denied in part for the reasons stated in the ''Director's Decision Under 10 CFR 2.206'' (DD–97–01), the complete text of which follows this notice. The decision and documents cited in the decision are available for public inspection and copying in the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, and at the local public document room located in the Van Wylen Library at Hope College in Holland, Michigan.

A copy of this decision has been filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c). As provided therein, this decision will become the final action of the Commission 25 days after issuance unless the Commission, on its own motion, institutes review of the decision within that time.

Dated at Rockville, MD., this 23d day of January 1997.

For the Nuclear Regulatory Commission.

Frank J. Miraglia,

*Acting Director, Office of Nuclear Reactor Regulation.*

### Director's Decision Under 10 CFR 2.206

*I. Introduction*

On September 19, 1995, the organizations Don't Waste Michigan and Lake Michigan Federation (Petitioners) filed a Petition pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206) requesting that the U.S. Nuclear Regulatory Commission (NRC) (1) find that Consumers Power Company (licensee) violated NRC requirements related to unloading procedures for dry storage casks for spent nuclear fuel, (2) suspend the licensee's use of the general license provisions related to dry cask storage of spent nuclear fuel, (3) require a substantial penalty be paid by the licensee, and (4) conduct hearings related to unloading procedures for dry storage casks at Palisades.

On September 30, 1996, the Petitioners amended the Petition by including additional information in support of their position that the licensee did not have a workable unloading procedure before loading the 13 dry storage casks currently in the Palisades independent spent fuel storage installation (ISFSI).

The Petition has been referred to me pursuant to 10 CFR 2.206. The NRC letter dated October 24, 1995, to Dr. Sinclair and Mr. Skavroneck, on behalf of the Petitioners, acknowledged receipt of the Petition. Notice of receipt was published in the Federal Register on October 31, 1995 (60 FR 55388).

On the basis of the NRC staff's evaluation of the issues and for the reasons given below, the Petitioners' requests are granted in part and denied in part.

*II. Background*

NRC regulations contain a general license that authorizes nuclear power plants licensed by the NRC, such as Palisades, to store spent nuclear fuel at the reactor site in storage casks approved by the NRC. (See 10 CFR part 72, subpart K.) In regard to dry cask storage of spent nuclear fuel at Palisades, the licensee opted to use the VSC–24 Cask Storage System designed by Sierra Nuclear Corporation. The VSC–24 Cask Storage System was added to the list of NRC certified casks in May 1993 (58 FR 17948). The associated certificate of compliance, Certificate Number 1007, specifies the conditions for use of VSC–24 casks under the general license provisions of 10 CFR part 72. Section 1.1.2, ''Operating Procedures,'' in the certificate of compliance for the VSC–24 casks, requires that licensees prepare an operating procedure related to cask unloading. Specifically, the condition states

> Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The operating procedures suggested generically in the SAR (safety analysis report) are considered appropriate, as discussed in Section 11.0 of the SER (safety evaluation report), and should provide the basis for the user's written operating procedures. The following additional written procedures shall also be developed as part of the user operating procedures:
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> 1. A procedure shall be developed for cask unloading, assuming damaged fuel. If fuel needs to be removed from the multi-assembly sealed basket (MSB), either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of oxidized fuel and to prevent radiological exposure to personnel during this operation. This activity can be achieved by the use of the Swagelok valves, which permit a determination of the atmosphere within the MSB before the removal of the structural and shield lids. If the atmosphere within the MSB is helium, then operations should proceed normally, with fuel removal, either via the transfer cask or in the pool. However, if air is present within the MSB, then appropriate filters should be in place to permit the flushing of any potential airborne radioactive particulate from the MSB, via the Swagelok valves. This action will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection in the form of respirators or supplied air should be