Morrissey by April 26 at the address indicated in the notice.

Organizations or individuals may also submit statements for the record without testifying. Twenty (20) copies of such statements should be sent to the Acting Executive Secretary of the Advisory Council at the above address. Papers will be accepted and included in the record of the meeting if received on or before April 26, 1996.

Signed at Washington, DC this 18th day of April, 1996.

Olena Berg,

Assistant Secretary, Pension and Welfare Benefits Administration.

[FR Doc. 96–10069 Filed 4–23–96; 8:45 am]

BILLING CODE 4510–29–M

Pension and Welfare Benefits Administration

Working Group on Small and Medium-Sized Employer-Sponsored Plans; Advisory Council on Employee Welfare and Pension Benefits Plans; Notice of Meeting

Pursuant to the authority contained in Section 512 of the Employee Retirement Income Security Act of 1974 (ERISA), 29 U.S.C. 1142, a public meeting of the Working Group on Small and Medium-Sized Plans of the Advisory Council on Employee Welfare and Pension Benefit Plans will be held on May 7, 1996, in Room N–3437 B&C, U.S. Department of Labor Building, Third and Constitution Avenue, NW., Washington, DC 20210.

The purpose of the meeting, which will run from 9:30 a.m. to noon, is to work to formulate guidance for small and medium-sized plans in selecting plan service providers.

Members of the public are encouraged to file a written statement pertaining to any topic concerning ERISA by submitting 20 copies on or before April 26, 1996 to Sharon Morrissey, Acting Executive Secretary, ERISA Advisory Council, U.S. Department of Labor, Suite N–5677, 200 Constitution Avenue, NW., Washington, DC 20210. Individuals or representatives of organizations wishing to address the Working Group on Small and Medium-Sized Plans of the Advisory Council should forward their request to the Acting Executive Secretary or telephone (202) 218–8753. Oral presentations will be limited to ten minutes, but an extended statement may be submitted for the record. Individuals with disabilities, who need special accommodations, should contract Sharon Morrissey by April 26 at the address indicated in this notice.

Organizations or individuals may also submit statements for the record without testifying. Twenty (20) copies of such statements should be sent to the Acting Executive Secretary of the Advisory Council at the above address. Papers will be accepted and included in the record of the meeting if received on or before April 26, 1996.

Signed at Washington, DC, this 18th day of April 1996.

Olena Berg,

Assistant Secretary, Pension and Welfare Benefits Administration.

[FR Doc. 96–10070 Filed 4–23–96; 8:45 am]

BILLING CODE 4510–29–M

NATIONAL INSTITUTE FOR LITERACY

Agency Information Collection Activities

ACTION: Notice.

SUMMARY: In compliance with the Paperwork Reduction Act (44 U.S.C. 3501 *et Seq.*), this notice announces an Information Collection Request (ICR) by the NIFL. The ICR describes the nature of the information collection and its expected cost and burden.

DATES: Comments must be submitted on or before June 21, 1996.

FOR FURTHER INFORMATION CONTACT: Sondra Stein at (202) 632–1508 or e-mail: sstein@nifl.gov.

SUPPLEMENTARY INFORMATION:

Title: Application for State-Capacity Building Awards to state officials to develop and implement interagency Performance Measurement and Reporting Systems that foster continuous improvement in adult literacy and basic skills programs.

Abstract: The National Literacy Act of 1991 established the National Institute for Literacy and required that the Institute conduct basic and applied research and demonstrations on literacy, collect and disseminate information to Federal, State and local entities with respect to literacy; and improve and expand the system for delivery of literacy services. This form will be used by State officials, including Governors, State Education Agencies, State Workforce Development Councils, and State Literacy Resource Centers to apply for funding to develop and implement Interagency Performance Measurement and Reporting Systems. Evaluations to determine successful applicants will be made by a panel of literacy experts using the publishing criteria. The Institute will use this information to make a maximum of six cooperative agreement awards for a period of up to 2 years.

Burden Statement: The burden for this collection of information is estimated at 55 hours per response. This estimate includes the time needed to review instructions, complete the form, and review the collection of information.

Respondents: Governors, State Education Agencies, State Workforce Development Councils, and State Literacy Resource Centers.

Estimated Number of Respondents: 20.

Estimated Number of Responses Per Respondent: 1.

Estimated Total Annual Burden on Respondents: 1100 hours.

Frequency of Collection: One time. Send comments regarding the burden estimate or any other aspect of the information collection, including suggestions for reducing the burden to: Sondra Stein, National Institute for Literacy, 800 Connecticut Ave., NW., Suite 200, Washington, DC 20006.

Andrew J. Hartman,

Director, National Institute for Literacy.

[FR Doc. 96–10146 Filed 4–23–96; 8:45 am]

BILLING CODE 6055–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 30, 1996, through April 12, 1996. The last biweekly notice was published on April 10, 1996 (61 FR 15985).

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 Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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 May 24, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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, 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: March 20, 1996.

Description of amendment request: The licensee proposes to relocate Technical Specification (TS) 3.3.3.2, Movable Incore Detectors, to the Harris Nuclear Plant Core Operating Limits Report (COLR). Future changes to the relocated provisions will be evaluated in accordance with 10 CFR 50.59.

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

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

The proposed change will simplify the Technical Specifications, while implementing the recommendations of the Commission's Final Policy Statement on TS Improvements. The changes are administrative in nature and do not involve any modifications to plant equipment or affect plant operation. Since the TS provisions are being relocated to a licensee-controlled document, any future changes will be controlled under 10 CFR 50.59. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is a relocation of existing Technical Specification provisions. It does not involve any physical alterations to plant equipment or alter the method by which any safety-related system performs its function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not affect any Final Safety Analysis Report (FSAR) Chapter 15 accident analyses or have any impact on margin as defined in the Bases to the Technical Specifications. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Eugene V. Imbro

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County; Northeast Nuclear Energy Company, et al., Docket Nos. 50–245, 50–336, 50–423, Millstone Nuclear Power Station, Units 1, 2, and 3, New London County, Connecticut; and North Atlantic Energy Service Company, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: February 1, 1996

Description of amendment request: The amendment request would revise Section 6 ''Administrative Controls,'' of the Haddam Neck Plant, Millstone Unit Nos. 1, 2, and 3, and Seabrook Station, Unit 1 Technical Specifications to reflect several changes in organizational titles. The proposed changes are administrative title and editorial changes only.

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

* * * The proposed changes do not involve an SHC because the change would not:

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

No design basis accidents are affected by these proposed changes. The proposed changes are administrative and editorial in nature and are being proposed to reflect the recently announced organizational changes which will become effective on February 1, 1996. These changes include: insertion of the function Chief Nuclear Officer, in lieu of Executive Vice President—Nuclear; and establishment of a single point of operational direction for all five units in the position of the Vice President—Nuclear Operations. This individual is in lieu of the positions of Vice President—Haddam Neck, Senior Vice President—Millstone Station, and Executive Director—Nuclear Production. These latter positions have been eliminated; other changes are: the appointment of the Haddam Neck Plant Nuclear Unit Director as chairman of the Haddam Neck PORC [Plant Operations Review Committee]; promotion of the Shift Supervisor/Shift Superintendent to the position of Shift Manager; revising the titles of ''additional operator'' and ''auxiliary operator'' to ''nuclear systems operator''; modifying the phrase ''crewman'' to a gender neutral term ''crewperson'';

reassignment of the delivery of ISEG [Independent Safety Engineering Group] reports to the Senior Vice President—Nuclear Safety and Oversight; and a change to the title of the Seabrook Station Manager to Station Director. No safety systems are adversely affected by the proposed changes, and no failure modes are associated with the changes. Therefore, there is no impact on the probability of occurrence or the consequences of any accidents previously evaluated.

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

Because there are no changes in the way the plants are operated due to this administrative change, the potential for an unanalyzed accident is not created. There is no impact on plant response, and no new failure modes are introduced. These proposed administrative and editorial changes have no impact on safety limits or design basis accidents, and they have no potential to create a new or unanalyzed event.

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

The changes do not directly affect any protective boundaries nor do they impact the safety limits for the protective boundaries. These proposed changes are administrative and editorial in nature. Therefore, there can be no reduction in the margin of safety.

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

Local Public Document Room locations: For the Haddam Neck Plant, Russell Library, 123 Broad Street, Middletown, CT 06457; for Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360; for Seabrook Station, Unit No. 1, Exeter Public Library, Founders Park, Exeter, NH 03833.

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

NRC Project Director: Phillip F. McKee

Duke Power Company, et al., Docket No. 50–413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of amendment request: January 26, 1996.

Description of amendment request: The amendment would allow a one-time change to the Technical Specifications (TS) to allow operation of the containment purge ventilation system during Modes 3 and 4 during startup following the forthcoming Unit 1 steam generator replacement outage. This would alleviate respiratory hazards to personnel who would enter the containment to perform surveillances during Modes 4 and 3 of startup operations. Those hazards are expected to result from the thermal decomposition product gases evolving from the heatup of newly installed thermal insulation. Operation of the containment purge system to exhaust these gases would ensure that the air quality meets applicable standards for personnel safety.

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

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

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

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

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

(2) The activity does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

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

Margin of safety is associated with confidence in the ability of the fission product barriers (the fuel and fuel cladding, the Reactor Coolant System pressure boundary, and the containment) to limit the level of radiation doses to the public. The proposed operation of the VP System will occur at the end of an extended outage. The level of decay heat and activity in the reactor is very low compared to the level of decay heat and activity associated with full power operations. For this reason, the likelihood of damage to the fuel following a DBLOCA [design basis loss-of-coolant analysis] occurring during the proposed purging is reduced, as determined above. Both offsite doses and doses to the Control Room were found to be small compared to the limits of 10 CFR [Part] 100 and GDC [General Design Criterion] 19. For these reasons, the activity does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: December 12, 1995.

Description of amendment request: The proposed amendments would correct an error in the Axial Flux Difference (AFD) Equations to more accurately reflect the proper AFD limit reduction, which is more conservative than the literal interpretation of the current Technical Specifications.

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

A. The change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The monitoring of core power distribution and peaking factors is to ensure accident analysis assumptions such as maximum local pin power at the initiation of an accident are satisfied, and are not involved in the initiation or mitigation of any previously evaluated accident.

The proposed change is actually more conservative than the existing Technical Specification currently being used at McGuire.

B. The change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

No plant modifications (hardware or control methods) are involved with this proposed change. The change is simply to correct an error in the Specification introduced in Amendments 130 (Unit 1) and 112 (Unit 2). The proposed change is more restrictive than the current specification. No changes are proposed which could create any new accident scenarios.

C. The proposed change will not involve a significant reduction in any margin of safety.

The proposed change ensures the margin of safety is properly maintained by properly reducing (instead of increasing) the Positive AFD [Axial Flux Difference] limit if a peaking factor exceeds its surveillance limit. The change is more conservative than the existing Specification and will ensure the margins of safety are properly maintained.

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: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: March 4, 1996.

Description of amendment request: The proposed amendments would delete the Flow Monitoring System from Technical Specification (TS) 3.4.6.1 and associated surveillance requirements. The TS requires that either the Containment Floor and Equipment Sump Level System or the Flow Monitoring System be used to ensure that Reactor Coolant leakage is maintained within the specified limits. Duke Power does not use the Flow Monitoring System as a result of documented instrumentation inaccuracies due to the as-built piping configuration. The existing piping configuration does not ensure a water solid line which is necessary for the correct operation of any type of flow instrumentation. Modification to add a loop seal downstream of the flow element would be necessary for operability, which would create access difficulties as well as increase the potential for a radiological hazard in the form of a CRUD trap.

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

1. This amendment will not significantly increase the probability or consequence of any accident previously evaluated.

This change will not increase the probability or consequences of an accident since this Reactor Coolant Leakage Detection instrumentation is not an accident initiator or mitigator.

This proposed Technical Specification change does not decrease the number of methods for Reactor Coolant leakage detection. This change will ensure there are still three distinctly separate methods of detecting NC [reactor coolant] leakage within the Containment Building. The first method will be detecting liquid leakage inside Containment via CFAE [Containment Floor and Equipment] level monitoring. The second method is detecting an increase in Radiation levels inside Containment and the third method is detecting steam leakage inside Containment. All three methods satisfy the diversity requirements listed in Regulatory Guide 1.45 for detecting a Reactor Coolant leak inside Containment.

The sensitivity requirement listed in Regulatory Guide 1.45 is to detect a Reactor Coolant leak of one (1) gpm in one (1) hour. The first method meets this by use of the Sump level monitoring and rate of increase alarm from this level monitoring device. There are two sumps inside containment and the levels for both sumps are combined for detecting a one (1) gpm leak. McGuire uses the Sump Level monitoring to adequately address liquid leakage detection inside Containment; therefore, a flow monitoring system on the Sump Discharge line is not necessary and can be deleted.

The Radiation Monitors are also set up to the required Regulatory Guide 1.45 sensitivity for detecting Reactor Coolant leakage and are not designed for SSE [safe-shutdown earthquake] events per the McGuire FSAR [Final Safety Analysis Report] (see McGuire's Request for Amendment: Reactor Coolant Leakage Detection Systems, dated March 4, 1996).

The third method for detecting Reactor Coolant leakage is to monitor Containment Ventilation Condensate Drain Tank (VUCDT) flow, for which McGuire is also using a level monitor. As in the case of the CFAE Unit Sump Level monitor, level monitoring for leakage detection is more reliable than flow monitoring.

2. This amendment will not create the possibility of any new or different kind of accident not previously evaluated.

The CFAE Flow Monitoring System has no control function, ([i.e.,] it is only a process monitor). Therefore, its deletion cannot create the pos[s]ibility of a new or different kind of accident.

3. This amendment will not involve a significant reduction in a margin of safety.

This proposed Tech Spec change does not decrease the number of methods for Reactor Coolant leakage detection. This change will ensure there are still three distinctly separate methods of detecting Reactor Coolant leakage within the Containment Building.

Tech Spec 3.4.6.1 specifies two Radiation Monitors as two separate required methods for Reactor Coolant Leakage Detection with the Containment Ventilation condensate level monitoring as a backup. The third method is the Containment Sump level monitoring with the flow monitoring as a backup.

The new standardized Tech Spec 3.4.15, lists method one as Containment Sump (Level OR Discharge Flow) Monitoring Device. McGuire proposes to use a Sump Level monitoring device only. The second method listed is one Containment Radiation Monitor (either the gaseous or particulate monitor). McGuire will still have both available. The third method listed is one Containment air cooler condensate flow rate monitor for which McGuire plans to also use a level monitor. Liquid, Radiation, and Steam monitoring will still be accounted for in the Tech Spec, with the additional requirement of running a Reactor Coolant leak calculation if any of the methods are inoperable.

Since McGuire is retaining three distinct methods of Reactor Coolant leakage detection per current TS [technical specification] requirements (and in agreement with current ISTS [improved standard technical specification] requirements), the proposed Technical Specification amendment does not cause any reduction in safety margin.

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: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

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

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: February 21, 1996.

Description of amendment request: The licensee proposes a change to the Plant Hatch Unit 1 and Unit 2 Technical Specifications. The proposed revision would change the Drywell Air Temperature Limiting Condition for Operation (LCO) from less than or equal to 135°F to less than or equal to 150°F. The proposed change would provide a margin for the primary containment Drywell Air Temperature LCO when prolonged summer and high river temperatures are experienced. Also, a correction to a Final Safety Analysis Report (FSAR) reference would be made. This typographical error is strictly editorial.

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The probability (frequency of occurrence) of previously evaluated accidents is not a function of the ambient drywell air temperature. Instrumentation setpoint calculations were assessed, and the increased ambient drywell air temperature does not affect any instrumentation setpoints or allowable values.

The design basis accidents were reevaluated utilizing the increased drywell air temperature as an initial assumption. The results indicated that no regulatory limits or equipment design requirements will be exceeded as the result of the proposed change. Therefore, the change in drywell air temperature does not result in a significant increase in the probability or consequences of any previously evaluated accidents.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed. Revising the Drywell Air Temperature LCO does not physically modify the plant nor does it modify the operation of any existing equipment.

3. The proposed change does not involve a significant reduction in a margin of safety. Design bases analyses performed utilizing 150°F as the initial drywell temperature demonstrate that design and regulatory limits are not exceeded. Equipment in the drywell required to mitigate the effects of a DBA [design basis accident] is qualified to operate under environmental conditions expected for an accident. Analysis results do not affect instrumentation setpoints or calibration, or accident equipment qualification.

Equipment qualified life is evaluated by an existing program which uses elevation-dependent drywell temperature rather than bulk average temperature. Therefore, the margin of safety associated with safety and other limits identified in the Technical Specifications are not significantly reduced.

The correction to an FSAR reference is strictly editorial. Therefore, it meets the three criteria stated above.

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. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: March 28, 1996 (TSCR 234).

Description of amendment request: The proposed amendment modifies statements in the Technical Specifications and bases to correctly reflect the reference parameter for anticipatory scram signal bypass.

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

1. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence or consequences of an accident.

This change modifies the terminology in a footnote to a Technical Specification Table and the bases. The change properly aligns the footnote and the bases with the FSAR [final safety analysis report] and the newly revised conservative setpoint which now correctly correlates the high pressure turbine third stage extraction steam line pressure to rated reactor thermal power. The change does not modify the function or operation of the bypass logic. Therefore, the proposed change will not increase the probability of occurrence or consequences of an accident.

2. State the basis for the determination that the activity does or does not create the possibility of an accident or malfunction of equipment of

a different type than any previously identified in the SAR.

The change does not involve any hardware and does not alter the functional intent of the pressure switches. The change of the footnote wording and the bases are primarily administrative and the existing Technical Specification Limiting Condition for Operation are preserved. Thus the proposed activity does not create the possibility of an accident or malfunction of a different type than any previously identified in the SAR.

3. State the basis for the determination that the margin of safety as defined in the bases of any Technical Specification is not reduced.

The revised setpoint assures that the anticipatory scram signal bypass is removed before reaching the Technical Specification limit of 40 percent rated reactor thermal power (during power ascension). Thus, the margin of safety as stated in the bases of Technical Specification 3.1 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 amendment request involves no significant hazards consideration.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: February 22, 1996 (U–602554)

Description of amendment request: The proposed amendment would modify Technical Specifications 3.3.8.1, ''Loss of Power Instrumentation,'' and 3.8.1, ''AC Sources-Operating.'' The proposed changes would delete the Surveillance Requirement (SR) 3.3.8.1.1 which requires a channel check for Loss of Power instrumentation and change Technical Specification Table 3.3.8.1–1 to change the allowable value for the Degraded Voltage Function (items 1.c and 2.c) from ''[greater than or equal to] 3762V and [less than or equal to] 3832V'' to ''[greater than or equal to] 3876V.'' The amendment would also change Technical Specification Table 3.3.8–1 to modify the Division 3 degraded voltage logic to be the same as Divisions 1 and 2 (i.e., two-out-of-two rather than three-out-of-three), and increase the steady state voltage from [greater than or equal to] 3740V to [greater than or equal to] 3870V for SRs 3.8.1.2, 3.8.1.7, 3.8.1.11, 3.8.1.12, 3.8.1.15, 3.8.1.19 and 3.8.1.20.

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

(1) None of the proposed changes involve a significant increase in the probability or consequences of any accident previously evaluated. Each of the proposed changes is evaluated against this criteria as discussed below.

The deletion of the channel check surveillance will result in discontinuing the recording of information that is not effective in assessing the capability of the degraded voltage relays to perform their intended function. Deletion of the channel check does not change the design or the expected performance of the Loss of Power (LOP) degraded voltage instrumentation, and therefore, the proposed change does not impact the intended function of this instrumentation to ensure adequate voltage for the ECCS equipment during DBA and other non-accident scenarios. This surveillance provides little added assurance of relay operability since the relay is normally in a ''non-tripped'' state.

The revision of the Allowable Values for the LOP degraded voltage and increase in the minimum required voltage for testing diesel generators will not result in any increase in the probability or consequences of any accident. The revised Allowable Values will continue to provide assurance that adequate voltage is available to run ECCS equipment during DBAs or any other non accident scenarios. With the emergency bus(es) voltage at or greater than the revised Allowable Values, the operability of required ECCS equipment is assured. The revised setpoints for the degraded voltage instrumentation, as controlled under 10CFR50.59 in the Clinton Power Station Operational Requirements Manual (ORM), are sufficiently low to assure that the possibility of spurious trips is minimized.

The planned modification for Division 3 LOP degraded voltage sensor/relay logic will make Division 3 logic identical to the present designs for Division 1 and 2. The proposed design for Division 3 will not result in an increase in the probability of any accident because the proposed LOP Degraded Voltage logic for Division 3 will be identical to the proven design of Division 1 and 2. There will not be an increase in the consequences of an accident because the design of the LOP Degraded Voltage instrumentation will continue to ensure adequate voltage for ECCS equipment during any DBA and during non-accident scenarios.

The proposed footnotes merely assure that the proposed changes become effective upon installation of the corresponding plant modifications. Thus, these changes are purely administrative.

Chapter 15 of the Clinton Updated Safety Analysis Report (USAR) discusses the effects of anticipated process disturbances to determine their consequences and the capability of the plant to control or accommodate such events. Subsection 15.2.6 discusses loss of AC power, including loss of grid voltage. This discussion demonstrates that fuel design limits and reactor coolant pressure boundary design conditions are not exceeded. The proposed changes do not affect the discussion nor the conclusion of this evaluation.

(2) None of the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated. Each of the proposed changes is evaluated against this criterion as discussed below.

The proposed changes (deletion of the channel check, the revised Allowable Value for the LOP degraded voltage instrumentation, revision of the minimum required voltage for the diesel generator (DG) surveillance, and change of the number of required channels for Division 3) do not alter the intent or purpose of the degraded voltage instrumentation. The instrumentation will continue to function to protect the loads on the emergency bus by switching automatically to the on site power source when the voltage has been at a degraded condition for greater than the Allowable Value of the time delay. The LOP instrumentation provides a responsive actuation (trip) to an accident or scenario where the protection provided by this function prevents damage to ECCS equipment during undervoltage (degraded voltage) conditions on the emergency bus(es). Because the instrumentation will continue to function to ensure that the emergency bus voltage for all three divisions is sufficient for the proper operation of all class 1E equipment down to the 120 volt level, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The change in the lower voltage for the DG surveillances will not impact the way the surveillances are conducted because the DGs are run as close to the nominal voltage as possible. The lower voltage is a criterion for evaluating the surveillance and the revised lower voltage is adequate for its intended purpose.

(3) None of the proposed changes involve a significant reduction in a margin of safety. Each of the proposed changes is evaluated against this criterion as discussed below.

The proposed deletion of the channel check SR 3.3.8.1.1 will not result in any reduction of the margin of safety because the channel check is ineffective and the status of the channel will continue to be apparent to plant personnel because of information provided by other TS required surveillances. The margin of safety is provided by LOP instrumentation ensuring the emergency bus(es) have adequate voltage to support ECCS operability. The proposed revision of the Allowable Value for the LOP degraded voltage will provide assurance that emergency bus(es) voltage will be adequate for ECCS loads during DBA and other non-accident scenarios. These setpoints were determined based on revised voltage calculations and using an NRC-approved setpoint methodology. Thus, these changes will not involve any reduction of the margin of safety. The proposed revision of the number of required channels for Division 3 will not result in a reduction in a margin of safety because the proposed Division 3 LOP Degraded Voltage instrumentation logic will be the same as the proven design of Division 1 and 2. This modification will improve plant maintenance and training by making Divisions 1, 2 and 3 similar thereby enhancing plant performance and safety.

Similarly, the proposed revision of the lower voltage limit for voltage for the DG surveillances (SR 3.8.1.2, SR 3.8.1.7, SR 3.8.1.11, SR 3.8.1.12, SR 3.8.1.15, SR 3.8.1.19, and SR 3.8.1.20) will assure that the DGs will be capable of controlling voltage to a range that will be adequate for the loads on the bus. This value was determined using revised voltage calculations and is consistent with the proposed degraded voltage setpoints. None of the proposed changes will 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: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727.

Attorney for licensee: Leah Manning Stetener, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525.

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: February 22, 1996 (U–602551).

Description of amendment request: The proposed amendment would change Technical Specification 3.4.11, ''Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits,'' to incorporate specific P/T limits for the bottom head region of the reactor vessel, separate and apart from the core beltline region of the reactor vessel.

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

(1) The proposed change results in a specific pressure and temperature (P/T) limit curve for the bottom head during vessel pressure testing evolutions, while the P/T limits for the remaining balance of reactor pressure vessel regions are unchanged. The limits for the bottom head region, which are only applicable during vessel system pressure or leak testing, were developed consistent with Regulatory Guide 1.99, Revision 2; 10CFR50, Appendix G; ASME Section III, Appendix G; and Welding Research Council (WRC) Bulletin 175. Additionally, the proposed change does not result in a change to the way in which the hydrostatic pressure tests are performed. That is, conformance to the P/T limits specified in Technical Specification Figure 3.4.11–1 with the proposed bottom head P/T limits incorporated, will continue to provide protection against brittle fracture of the vessel system during required testing so that vessel integrity is maintained. Therefore, this proposed change does not result in an increase in the probability or consequences of any accident previously evaluated.

(2) The proposed change does not result in any change to the plant or the way in which the hydrostatic pressure tests are performed. As a result, no new failure modes are introduced. Therefore, the proposed change cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The new P/T limit curve for the bottom head has been developed consistent with Regulatory Guide 1.99, Revision 2; 10CFR50, Appendix G; ASME Section III, Appendix G; and Welding Research Council (WRC) Bulletin 175. All other regions of the reactor pressure vessel retain their applicability to appropriate and previously approved P/T limit curves which are based on the same methodology. Conformance to the P/T limit curves, with the proposed changes incorporated, will continue to provide adequate margins of safety against brittle fracture of the reactor vessel. Therefore, this proposed change does not result in a significant reduction in the margin of safety.

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

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

Attorney for licensee: Leah Manning Stetener, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525.

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: February 22, 1996 (U–602522)

Description of amendment request: The proposed amendment would change Technical Specification 3.3.4.1, ''End of Cycle Recirculation Pump Trip (EOC–RPT) Instrumentation,'' by deleting Surveillance Requirement (SR) 3.3.4.1.6. The SR requires the reactor recirculation pump trip breaker interruption time to be determined at least once per 60 months.

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

(1) End of cycle recirculation pump trip (EOC–RPT) actuation in response to main generator load rejection and main turbine trip events has previously been evaluated in Chapter 15 of Clinton Power Station (CPS) Updated Final

Safety Analysis. The proposed change does not affect the initiators of any of these events. In addition, the possibility of failure of the EOC–RPT breaker to mitigate these events has not been increased because there has been no change in design and no change to the plant. Deleting the requirement to periodically measure the breaker arc suppression time will not impact the EOC–RPT breakers' capability of performing their intended function because CPS will continue to perform inspections, testing and maintenance that supports breaker operation as intended and provides assurance that breaker interruption time will be within limits. Thus, the EOC–RPT breaker trip may be expected to operate as before to mitigate pressurization transient effects.

The EOC–RPT breaker trip is also assumed to occur in the analyses for the loss of feedwater heating, feedwater controller failure, pressure regulator failure, recirculation flow control failure, and recirculation pump seizure events. However, the EOC–RPT breaker trip is not an initiator or mitigating feature for these events. The proposed change cannot therefore impact the probability or consequences for these events. Nonetheless, the EOC–RPT breaker trip may be assumed to function as before for these scenarios.

For scenarios where the EOC–RPT breaker trip could initiate an event (i.e., inadvertent recirculation pump trip events), the probability of occurrence is not increased. The design and operation of the EOC–RPT system has not been changed, and therefore, the consequences resulting from the EOC–RPT breaker trip are unchanged.

Based on the above, neither the probability nor the consequences of any accident previously evaluated have been increased.

(2) As noted above, the EOC–RPT breakers will continue to function as before. The proposed change involves no design change or physical change in the plant. Therefore, previous accident analyses are unchanged. Further, no new operations or testing is involved. On this basis, no new failure modes are introduced. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) This proposed change does not involve a significant reduction in a margin of safety. The capability of the EOC–RPT breaker trip to provide additional insertion of negative reactivity for mitigating design-basis events remains unchanged. That is, the EOC–RPT will continue to be capable of reducing the peak reactor pressure and power resulting from turbine trip or generator load rejection transients, thus providing additional margin to core thermal MCPR Safety Limits.

The margin of safety is assured by the EOC–RPT breaker trip occurring within established limits such that the overall system performs its intended safety function within the time analyzed for the system safety response. No system time limit change is proposed. The robust design of the breakers, combined with continued performance of vendor-recommended testing and maintenance that ensures proper mechanical and electrical performance of the breakers, will continue to provide assurance that breaker interruption time is within the acceptable limit. Therefore, there is no significant reduction in the margin of safety.

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

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

Attorney for licensee: Leah Manning Stetener, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525.

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: February 22, 1996 (U–602549).

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.6.5.1, ''Drywell,'' to allow drywell bypass leakage tests to be performed at intervals of up to ten years based, in part, on the demonstrated performance of the drywell barrier with respect to leak tightness. The proposed amendment would also revise TS 3.6.5.2, ''Drywell Air Lock,'' to extend the testing intervals for the surveillances on drywell air lock overall leakage and interlock operability, relocate the specific leakage limits on the air lock barrel and door seals to the TS Bases, relocate the requirement to pressurize the drywell air lock to 19.7 psid prior to performance of the overall drywell air lock leakage test to the TS Bases, and other administrative changes.

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

(1) The proposed changes do not involve a change to the plant design or operation. As a result, the proposed changes do not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. Therefore, the proposed changes cannot increase the probability of any accidents previously evaluated.

The proposed changes do potentially affect the leaktight integrity of the drywell, a structure used to mitigate the consequences of a loss of coolant accident (LOCA). The function of the drywell is to force the steam released from a LOCA through the suppression pool, limiting the amount of steam released to the primary containment atmosphere. This serves to limit the containment pressurization due to the LOCA. The leakage of the drywell is limited to ensure that the primary containment does not exceed its design limits of 185°F and 15 psig. Because the proposed change to replace the current 18-month frequency for performing drywell bypass leakage tests (DBLRTs) with a performance-based frequency does not alter the plant design, the proposed change does not directly result in an increase in the drywell leakage. However, decreasing the test frequency can increase the probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. This potential has been evaluated, and Illinois Power has determined that the proposed change to the DBLRT frequency will not result in the potential for undetected, large increases in leakage, as further discussed below.

There are several potential drywell bypass leakage paths. These include potential cracks in drywell concrete structure, the drywell vacuum breakers, and various penetrations through the drywell structure. Based on the results of the structural integrity test conducted at the design pressure of 30 psig as part of the preoperational test program, additional cracking of the drywell is not expected during the remaining life of the plant. Ventilation and piping penetrations (including the drywell vacuum breaker penetrations) are designed to ASME Code Class 2 and Seismic Category 1 requirements. These penetrations are typically designed with two isolation valves in series with one valve in the drywell and another either outside primary containment or in the wetwell. Technical Specification (TS) Surveillance Requirements (SRs) require, as applicable, periodic verification of drywell isolation valve position, stroke time, and automatic isolation capability. High energy lines that extend into the wetwell, such as the main steam lines and feedwater lines, are encapsulated by guard pipes to direct energy back into the drywell in case of a piping rupture. Electrical penetrations are sealed with a high strength/density material that will prevent leakage, as well as provide radiation shielding.

The proposed changes for the drywell air lock involve relocation of the separate limits on the drywell air lock barrel and seal leakage rates to the TS Bases, relocation of the requirement to pressurize the air lock to 19.7 psid prior to performance of the air lock overall (barrel) leakage test, and changing the frequency for these tests from 18 months to 24 months. While the proposed changes will eliminate separate TS limits on leakage of the drywell air lock, the overall drywell bypass leakage TS limit (which includes leakage through the air lock) is not affected by this proposed change. The limiting scenario for drywell bypass leakage is a small break LOCA which results in drywell pressures of approximately 3 psid. Only a large break LOCA can create drywell pressures of 19.7 psid. For this event, the allowable drywell bypass leakage rate is over eight times larger than for a small break LOCA. Thus, relocation of these requirements to the TS Bases will continue to provide adequate control of these requirements. The proposed air lock overall leakage rate testing frequency is consistent with the guidance for testing primary containment air locks in Nuclear Energy Institute (NEI) 94–01, ''Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J.'' The drywell air lock is tested in a manner similar to the primary containment air locks, even though the drywell air lock is not a direct leakage path from primary containment and, therefore, 10CFR50, Appendix J test requirements do not apply. The drywell air lock's use is limited during plant operation due to radiation and temperature in the drywell. Since sufficient confidence in the door's sealing capability is assured considering past performance and the air lock door usage is very low throughout an operating cycle, it is justified to allow performance of these tests at refueling-outage intervals, whether the unit is on a 18-month or a 24-month refueling cycle.

Operational experience has shown that the leak tightness of the drywell has been maintained well below the allowable leakage limits at Clinton Power Station. The TS limit of 10% of the design [maximum allowable leakage path area] provides a large margin for degradation. Drywell performance to date suggests that drywell degradation, even with a ten-year interval between tests, will not exceed this margin. The most recent DBLRT performed during the fourth refueling outage (RF–4) measured a drywell bypass leakage rate of 0.07% of the design limit.

An analysis was also conducted to determine the potential risk to the public from unacceptable drywell bypass leakage going undetected as a result of the proposed change. Based on this probabilistic risk analysis, for several different accident scenarios, the risk of radioactivity release from containment was found to be insignificant.

Based on the above, Illinois Power has concluded that the proposed changes will not result in a significant increase in the consequences of any accident previously evaluated.

(2) The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. Drywell bypass leakage cannot, of itself, create an accident. Thus, it has been concluded that the proposed change cannot create the possibility of an accident not previously evaluated.

(3) The NRC has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(c). These proposed changes involve the withdrawal of operating restrictions previously imposed because acceptable operation of the Mark III primary containment design had not been demonstrated at the time of initial licensing. As published in the Federal Register (FR) regarding no significant hazards consideration criteria, granting of a relief based upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation had not yet been demonstrated does not involve a significant hazards consideration (reference 48 FR 14870).

The proposed change only affects the frequency of measuring the drywell bypass leakage rate and does not change the bypass leakage rate limit. The proposed change could potentially increase the probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. However, operational experience has shown that the leaktightness of the drywell has been maintained well below the allowable leakage limits. In addition, there are TS surveillances which require, as applicable, periodic verification of drywell isolation valve position, stroke time, and automatic isolation capability. Further, qualitative methods (such as periodic verification that the drywell pressurizes, which ensures that the drywell leak rate is less than the instrument air leak and usage rates) are available to provide assurance that the drywell leakage rate is being maintained within limits. The Clinton Power Station TS require the drywell leakage rate measured during DBLRTs to be less than or equal to 10% of the design limit. This request does not affect this required margin. Nor does it affect the existing margin between the primary containment design pressure and the actual pressure at which primary containment would fail.

With respect to proposed changes to the drywell air lock overall leakage testing and interlock testing requirements, the proposed leak test frequencies are consistent with the guidance for testing primary containment air locks in NEI 94–01. Due to the limited use of the drywell air locks during plant operation, it is justified to allow performance of interlock operability testing on a refueling outage basis, whether the unit is on an 18-month or a 24-month refueling cycle. The separate limits on the drywell air lock and barrel are being relocated from the TS, these limits are being controlled under 10CFR50.59 and the TS Bases Control program of TS 5.5.11. Leakage through these pathways will continue to be a part of the overall drywell bypass leakage limited by LCO 3.6.5.1.

An analysis was also conducted to determine the potential risk to the public from the proposed change. Based on this probabilistic risk analysis, for several different accident scenarios, the risk of radioactivity release from containment was found to be insignificant.

As a result, Illinois Power has concluded that the proposed changes will continue to assure that the drywell bypass leakage will be within design limits if challenged and therefore, will not result in a significant reduction in the margin of safety.

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

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

Attorney for licensee: Leah Manning Stetener, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525.

NRC Project Director: Gail H. Marcus.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of amendment requests: February 26, 1996 (AEP:NRC:1071U).

Description of amendment requests: The proposed amendments would modify the technical specifications (TS) to increase the current limit on nominal fuel assembly enrichment for new, Westinghouse-fabricated, fuel stored in the new fuel storage racks from 4.55 weight percent uranium-235 isotope to 4.95 weight percent uranium-235 isotope with certain provisions. Also, TS 5.6.2 would be reformatted similar to that used in the standard TS (NUREG–1431, Rev. 1).

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

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,

(2) create the possibility of a new or different kind of accident from any accident previously evaluated, or

(3) involve a significant reduction in a margin of safety.

Criterion 1

The proposed changes will not involve a significant increase in the probability of an accident previously evaluated because similar administrative controls to those presently used to identify new fuel storage rack inventory and compliance with T/S limits will be used. There are no physical changes to the plant associated with this T/S change. The consequences of an accident previously evaluated will not be increased because the reactivity of the fuel stored in the new fuel storage racks under the proposed T/S limits will be no greater than the reactivity of fuel stored in the new fuel storage racks presently allowable under the current T/S limits.

Criterion 2

The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because the changes will involve no physical changes to the plant nor any changes in plant operations. Furthermore, the reactivity of the fuel stored in the new fuel storage racks under the proposed T/S limits will be no greater than the reactivity of fuel stored in the new fuel storage rack presently allowable under the current T/S limits.

Criterion 3

The proposed amendment will not involve a significant reduction in a margin of safety because the reactivity of the fuel stored in the new fuel storage racks under the proposed T/S limits will be no greater than the reactivity of fuel stored in the new fuel storage racks presently allowable under the current T/S limits.

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. In addition, the reformatting of TS 5.6.2 is a purely administratiave change having no effect on the physical plant or its operation. Therefore, the NRC staff proposes to determine that the amendment requests involve 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: Mark Reinhart, Acting.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of amendment requests: February 29, 1996 (AEP:NRC:1232).

Description of amendment requests: The proposed amendments would revise the technical specifications to reduce the boric acid concentration in the boric acid storage system from approximately 12 percent to approximately 4 percent by weight. Related changes are also proposed to increase the minimum required flow rate in action statements for certain affected TS and add an additional surveillance requirement for this flow rate, and decrease the minimum temperature requirement in certain affected TS to 63 °F. The bases section is also updated to reflect these proposed changes.

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

Per 10 CFR 50.92, a proposed change does not involve a significant hazards consideration if the change does not:

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

2. create the possibility of a new or different kind of accident from any accident previously evaluated, and

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

Criterion 1

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

NO. The BAST [boric acid storage tank] water volume and boron concentration were not credited in any Chapter 14 safety analysis. Therefore, no change in the probabilities of the accident analysis will result from the BAST water volume and boron concentration change. In addition, since the BAST water volume and boron concentration are not taken into consideration in any safety analysis, the consequences of an accident previously evaluated in the FSAR [final safety analysis report] are not increased. The heat tracing system is currently only necessary to prevent precipitation of existing high boric acid concentration in the plant systems. The reduction in boron concentration in this proposal eliminates the need for the heat tracing system. The existence of the heat tracing system was not part of any safety analysis and disabling of the heat tracing system will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

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

NO. Since the minimum required water flow from the boric acid storage system to the reactor coolant system was increased to counteract any possible operational transients, as shown in Attachment 4 [of the application], the change in BAST water volume and boron concentration and disabling of the heat tracing system do not create the possibility of an accident which is different from any already evaluated in the FSAR. No new or different failure modes have been defined for any system or component nor has any new limiting single failure been identified.

Criterion 3

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

NO. The margin of safety requirements are not affected by the removal of the heat tracing system and the reduction of the boric acid concentration in the boric acid storage system. The required flow paths and borated water sources are unaffected by this proposal. The required quantity of borated water is still available based upon the performed evaluation, and appropriate surveillance requirements ensure the ability to deliver this borated water. The reduction of the boric acid concentration in the BASTs will ensure that the boric acid remains in solution at the normal room temperature in the auxiliary building. With the above changes, there will be a net improvement in system reliability and accordingly the proposed changes do not affect 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 requests involve 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: Mark Reinhart, Acting.

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

Date of amendment requests: March 13, 1996.

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise TS 4.0.5, ''Surveillance Requirements,'' to delete reference to prior NRC approval for written relief from the Inservice Inspection (ISI) and Inservice Testing Program (IST) requirements and to add ASME Section XI definition of ''Biennially or every 2 years—At least once per 731 days'' in TS 4.0.5b.

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

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

The proposed changes implement the NRC's recommendation contained in NUREG–1482, ''Guidelines for Inservice Testing Programs at Nuclear Power Plants,'' endorsed by Generic Letter 89–04, Supplement 1, ''Guidance on Developing Acceptable Inservice Testing Programs.'' The changes are consistent with 10 CFR 50.55a, ''Codes and Standards,'' which does not prohibit the implementation of relief from ASME Section XI requirements prior to specific written approval when those changes are found acceptable by change process specified in 10 CFR 50.59, ''Changes, Tests and Experiments.'' The proposed changes are administrative in nature and do not involve any modifications to any plant equipment or affect plant operation.

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

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 are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety-related system performs its function.

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

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

The proposed changes do not alter the basic regulatory requirements and do not affect any safety analyses.

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

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

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: April 3, 1996.

Description of amendment request: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise Technical Specifications 3/4.7.5, ''Control Room Ventilation System,'' 3/4.7.6, ''Auxiliary Building Safeguards Air Filtration System,'' and 3/4.9.12, ''Fuel Handling Building Ventilation System,'' to clarify the testing methodology utilized by PG&E to determine the operability of the charcoal and high-efficiency particulate air (HEPA) filters in the engineering safeguards features (ESF) air handling units.

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

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

The charcoal testing protocol changes will not affect system operation or performance, nor do they affect the probability of any event initiators. These changes do not affect any engineered safety features actuation setpoints or accident mitigation capabilities. The new charcoal adsorber sample laboratory testing protocol more accurately demonstrates the required performance of the adsorbers in the control room ventilation system and auxiliary building safeguards air filtration system following a design basis loss of coolant accident or in the fuel handling building ventilation system following a fuel handling accident outside containment. The decontamination efficiencies used in the offsite and control room dose analyses are not affected by these changes. Therefore, offsite and control room dose analyses are not affected by this change, and all offsite and control room doses will remain within the limits of 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criterion (GDC) 19.

The requirements of ANSI N510–1980 encompass the requirements of ANSI N510–1975, which is referenced in Regulatory Guide (RG) 1.52, as it applies to testing at Diablo Canyon Power Plant (DCPP). Consequently, revising the Technical Specifications (TS) to reference ANSI N510–1980 will have no effect on filter testing.

The proposed changes are consistent with the new Standard Technical Specifications (NUREG–1431, Rev. 1).

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

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

The changes to the charcoal sample testing protocol will not affect the method of operation of the system. The proposed changes only affect the testing parameters for the charcoal samples. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes.

The requirements of ANSI N510–1980 encompass the requirements of ANSI N510–1975, which is referenced in RG 1.52, as it applies to testing at DCPP. Consequently, revising the TSs to reference ANSI N510–1980 will have no effect on filter testing.

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

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

The changes in charcoal sample testing protocol will not affect system performance or operation. The decontamination efficiencies used in the offsite and control room dose analyses are not affected by these changes. Therefore, offsite and control room dose analyses are not affected by this change, and all offsite and control room doses will remain within the limits of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19.

The requirements of ANSI N510–1980 encompass the requirements of ANSI N510–1975, which is referenced in RG 1.52, as it applies to testing at DCPP. Consequently, revising the TSs to reference ANSI N510–1980 will have no effect on filter testing.

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

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

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: March 13, 1996.

Description of amendment request: The proposed amendment would revise the Humboldt Bay Power Plant (HBPP), Unit 3, Technical Specifications (TS) by incorporating position changes to reflect a proposed plant staff reorganization. The TS changes proposed are as follows:

(1) TS Section VII.C.2.c and VII.D.1.b—change the position title from ''Power Plant Engineer'' to ''Senior Power Production Engineer.''

(2) TS Section VII.C.2.d—change the position title from ''Senior Chemical and Radiological Engineer'' to ''Senior Radiation Protection Engineer.''

(3) TS Section VII.C.2.e and VII.D.1.b—change the position title from ''Maintenance Planner'' to ''Supervisor of Maintenance.''

(4) TS Section VII.C.2.g and VII.D.1.b—add the position of ''Assistant Plant Manager/Power Plant Engineer.''

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

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

The proposed administrative and organizational changes provide editorial corrections and reflect the proposed HBPP and current NRC organizations. These changes do not affect the operating methodology of HBPP, and they are not related to the probability or consequences of an accident previously evaluated.

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

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

The proposed revisions to the HBPP TS are organizational and administrative in nature, and do not change the method by which any safety-related system performs its function.

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

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

The proposed changes have no effect on the current operating methodologies or actions that govern plant performance. In addition, the proposed changes do not affect the margin of safety associated with parameters for any accident analysis.

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

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

Local Public Document Room location: Humboldt County Library, 1313 3rd Street, Eureka, California 95501.

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

NRC Project Director: Seymour H. Weiss.

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

Date of application request: February 9, 1996, as superseded by letter dated March 22, 1996.

Description of amendment request: The amendment would revise Technical Specification (TS) Definition 1.7, TS 3/4.6, TS 6.8, and their associated bases to directly reference Regulatory Guide 1.163 as required by 10 CFR 50, Appendix J, Option B, for the Type A containment integrated leak rate tests (ILRTs) and the Type B and C local leak rate tests (LLRTs).

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

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

The proposed changes to TS 1.7e, 4.6.1.1, 3/4.6.1.3, Bases 3/4.6.1.1 and the program addition to TS 6.8.4g have no effect on plant operation. The proposed changes only provide mechanisms within TS for implementing a performance-based methodology for determining the frequency of leak rate testing, as allowed by the NRC. The test type, method, and acceptance criteria will not be changed. Containment leakage will continue to be maintained within the required limits. Based on industry and NRC evaluations performed in support of developing Option B, these changes potentially result in a minor increase in the consequences of an accident previously evaluated due to the increased testing intervals. However, the proposed changes do not result in an increase in the core damage frequency since the containment system is used for mitigation purposes only.

Directly referencing the Containment Leakage Rate Testing Program for Containment ILRT and LLRT requirements does not involve any modification to plant equipment or affect the operation or design basis of the containment. Leakage rate testing is not a precursor to or an initiating event for any accident.

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

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

The proposed changes only allow for implementation of 10 CFR 50, Appendix J, Option B and do not involve any modifications to any plant equipment or affect the operation or design basis of the containment. The proposed changes do not affect the response of the containment during a design basis accident.

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

The proposed changes do not affect or change a safety limit, any limiting condition for operation or affect plant operations. The changes only implement the Appendix J, Option B test frequencies that have been determined by NRC not to involve a safety concern. The testing methods, acceptance criteria and bases are not changed and still provide assurance that the containment will provide its intended function.

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

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: March 21, 1996.

Description of amendment request: The proposed changes to the Technical Specifications (TS) for the North Anna Power Station, Units 1&2 (NA–1&2) would clarify the requirements for testing charcoal adsorbent in the Waste Gas Charcoal Filter System, the Control Room Emergency Habitability System, and the Safeguards Area Ventilation System. No change in the testing is being proposed, only clarification of the description of the required testing in TS 3/4.6.4.3, 3/4.7.7.1, and 3/4.7.8.1.

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

The proposed Technical Specifications changes will revise Surveillance Requirements for the charcoal adsorbent in the Waste Gas Charcoal Filter System (TS 3/4.6.3.), Control Room Emergency Habitability System (TS 3/4.7.7.2), and the Safeguards Area Ventilation System (TS 3/4.7.8.1) to reflect the current testing methodology for new and used carbon adsorbent. These proposed changes specify ASTM D 3803–1979 as the laboratory testing standard for both new and used charcoal adsorbent for the ventilation system identified above.

Virginia Electric and Power has evaluated the proposed Technical Specification changes to the North Anna Units 1 and 2 Technical Specifications against the Significant Hazards Criteria of 10 CFR 50.92 and determined that the changes do not involve any significant hazard for the following reasons:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

The proposed changes are administrative in nature in that the changes only explicitly specify the current testing methodology for charcoal adsorbent. The proposed changes will not affect system operation or performance, nor do they affect the probability of any event initiators. These changes do not affect any Engineered Safety Features actuation setpoints or accident mitigation capabilities. Therefore, the proposed changes will not significantly increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

The proposed changes only clarify the requirements for charcoal testing and will not affect the method of operation of the ventilation systems. Furthermore, the proposed changes are only intended to clarify the existing requirements to explicitly specify the current test methodology. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

3. The margin of safety has not been significantly reduced.

The proposed changes which represent the current laboratory testing methodology for charcoal adsorber samples, demonstrates the required performance of the adsorbent following a design basis LOCA or Fuel Handling Accident. Changing the Technical Specification to clarify the methodology for charcoal sample testing will not affect system performance or operation.

Therefore, these changes will not result in a significant reduction in any margin of safety.

Based on the above discussions, it has been determined that the requested Technical Specification changes do not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluation; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Eugene V. Imbro.

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

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

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

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

Date of amendment request: March 21, 1996.

Brief description of amendments: The amendments provide changes to Technical Specifications (TS) for CR3 relating to the Once Through Steam Generator's (OTSG's) tube inspection acceptance criteria, and repair limit for removing steam generator tubes from service. The proposed TS change would be applicable for one cycle duration, and only to Inter-Granular-Attack (IGA) degradation mechanism in a limited region of the OTSG.

Date of publication of individual notice in Federal Register: March 28, 1996 (61 FR 13888)

Expiration date of individual notice: April 29, 1996.

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: May 30, 1995, as supplemented by letter dated February 8, 1996.

Description of amendment request: The proposed amendment would increase the spent fuel pool heat load licensing basis to provide greater flexibility for normal refueling practices.

Date of individual notice in the Federal Register: April 3, 1996 (61 FR 14832)

Expiration date of individual notice: May 3, 1996.

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

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.

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

Date of application for amendments: December 20, 1995.

Brief description of amendments: These amendments change the instrument setpoint for the reactor trip and main steam isolation signal actuation on low steam generator pressure from greater than or equal to 919 psia with an allowable value of 911 psia to 895 psia with an allowable value of greater than or equal to 890 psia.

Date of issuance: April 5, 1996.

Effective date: April 5, 1996, to be implemented within 45 days of issuance.

Amendment Nos.: Unit 1–105; Unit 2–97; Unit 3–77.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7544) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 5, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 1, 1995 as supplemented on December 1, 1995.

Brief description of amendments: The amendments reflect the new plant electrical distribution configuration, surveillance and limiting condition for operation of the new safety-related (SR) emergency diesel generator (EDG), the increased electrical capacities for the two of the three existing SR EDGs, the increased EDG fuel oil storage capacity, and the fire protection system for the new EDG building. The remaining existing SR EDG will be upgraded during the Unit No. 2 refueling outage scheduled for the spring of 1997.

Date of issuance: April 2, 1996.

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

Amendment Nos.: 214 and 191.

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

Date of initial notice in Federal Register: January 3, 1996 (61 FR 175) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated April 2, 1996.

No significant hazards consideration comments received: No.

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

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

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: December 6, 1995, as supplemented February 27, 1996, and March 28, 1996.

Brief description of amendments: The amendments modify the technical specifications to replace the existing scheduling requirements for overall integrated and local containment leakage rate testing with a requirement to perform the testing in accordance with 10 CFR Part 50, Appendix J, Option B. Option B allows test scheduling to be adjusted based on past performance.

Date of issuance: April 4, 1996.

Effective date: April 4, 1996.

Amendment Nos.: 81, 81, 73, and 73.

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7547) The February 27, 1996, and March 28, 1996, supplements modified the Technical Specification pages to be more consistent with the published guidance, were within this scope of the initial notice, and did not affect the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 1996.

No significant hazards consideration comments received: No.

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

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

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: October 3, 1995, as supplemented on February 21, 1996, and April 2, 1996.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to implement ten of the line-item TS improvements recommended in Generic Letter (GL) 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993. The amendments also include editorial changes on the affected TS pages.

Date of issuance: April 10, 1996.

Effective date: April 10, 1996.

Amendment Nos.: 82, 82 and 74, 74.

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

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58397). The February 21, 1996, and April 2, 1996, submittals did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 10, 1996.

No significant hazards consideration comments received: No.

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

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

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: May 17, 1995, as supplemented by letters dated January 17, March 8, March 18, April 4 and April 9, 1996.

Brief description of amendments: The amendments revised the Facility Operating Licenses and the technical specifications to permit the steam generator tubes to be repaired using the tungsten inert gas welded sleeve process developed by ABB-Combustion Engineering and remove references to the kinetically welded sleeving process.

Date of issuance: April 12, 1996.

Effective date: April 12, 1996.

Amendment Nos.: 83, 83, 75, and 75.

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

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35064) The additional submittals provided information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 12, 1996.

No significant hazards consideration comments received: No.

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

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

Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: September 1, 1995, for Dresden and September 20, 1995, for Quad Cities.

Brief description of amendments: This application upgrades the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG–0123, ''Standard Technical Specification General Electric Plants BWR/4.'' This application upgrades only Section 6.0, ''Administrative Controls.''

Date of issuance: April 2, 1996.

Effective date: Immediately, to be implemented no later than June 30, 1996.

Amendment Nos.: 149, 143, 170, and 166.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 20, 1995 (60 FR 48728) for Dresden and October 5, 1995 (60 FR 52226) for Quad Cities. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 2, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 18, 1996, as supplemented on March 1, March 22, March 26, and April 3, 1996.

Brief description of amendments: The amendments change the setpoints for the automatic primary containment isolation signal upon detection of a high main steamline tunnel differential temperature and delete the automatic isolation function upon detection of a high main steamline tunnel temperature. Additionally, the amendments provide a 12 hour allowed outage time for the Main Steam Line Tunnel Differential Temperature—High isolation signal upon loss of the Reactor Building Ventilation System.

Date of issuance: April 4, 1996.

Effective date: Immediately, to be implemented prior to restart from refueling outage L1R07 (Unit 1) and L2R07 (Unit 2).

Amendment Nos.: 111 and 96.

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

Date of initial notice in Federal Register: February 27, 1996 (61 FR 7281). The March 1, March 22, March 26 and April 3, 1996, submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 25, 1995 as supplemented on December 15, 1995, February 5, February 9, February 28, March 4, March 28 and April 3, 1996.

Brief description of amendments: These amendments revise the LaSalle Facility Operating Licenses and Technical Specifications (TSs) to reflect the deletion of the leakage control system (LCS) presently installed to control and contain the leakage past the main steamline isolation valves (MSIVs) on each of the four main steamlines. The TSs are also revised to raise the allowable leakage rates from 25 standard cubic feet per hour (scfh) for each set of MSIVs and a total of 100 scfh from all four main steamlines to values of 100 scfh per steamline and 400 scfh for all four steamlines.

Date of issuance: April 5, 1996.

Effective date: Immediately, to be implemented by startup from refueling outage L1R07 (Unit 1) and L2R07 (Unit 2).

Amendment Nos.: 112 and 97.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the licenses and technical specifications.

Date of initial notice in Federal Register: October 25, 1995 (60 FR 54717). The December 15, 1995, February 5, February 9, February 28, March 4, March 28 and April 3, 1996, submittals provided additional information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 5, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 16, 1994, as supplemented February 6, 1995.

Brief description of amendment: The amendment revises License Condition 2.K and relocates the Indian Point Nuclear Generating Unit No. 2 (IP2) fire protection requirements from the IP2 Technical Specifications to the IP2 fire protection program plan in accordance with the guidance provided in Generic Letter (GL) 86–10, ''Implementation of Fire Protection Requirements,'' April 24, 1986, and GL 88–12, ''Removal of Fire Protection Requirements from Technical Specifications,'' August 2, 1988.

Date of issuance: March 26, 1996.

Effective date: As of the date of issuance to be implemented within 9 months.

Amendment No.: 186.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications and the Facility Operating License.

Date of initial notice in Federal Register: August 17, 1994 (59 FR 42335) The February 6, 1995, submittal provided clarifying information and did not expand the scope of the original application, and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 17, 1995.

Brief description of amendment: This amendment revises the Palisades Facility Operating License to reference 10 CFR Part 40, allow the use of source materials as reactor fuel, delete references to specific amendments and specific revisions in the listed titles of the Physical Security Plan, Suitability Training and Qualification Plan, and the Safeguards Contingency Plan and make minor editorial changes to the license. In addition, the Technical Specifications (TS) are modified as follows: (1) TS 3.1.2 is modified to change the pressurizer cooldown limit from 100°F to 200°F/hour; (2) the shield cooling system requirements are relocated to the Final Safety Analysis Report; (3) several minor editorial changes and corrections are made, including corrections requested in the licensee's letter of March 24, 1995; and (4) several TS bases pages have been revised. The portion of the amendment request deleting license paragraph 2.F on reporting requirements was denied.

Date of issuance: April 5, 1996.

Effective date: April 5, 1996.

Amendment No.: 171.

Facility Operating License No. DPR–20: Amendment revised the Facility Operating License and the Technical Specifications.

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58399).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 5, 1996, and an Environmental Assessment dated March 11, 1996 (61 FR 10811).

No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: December 7, 1995, as supplemented January 4, March 1, March 5, March 7, March 11, March 27, and March 29, 1996.

Brief description of amendment: The amendment revises Technical Specifications 3/4.4.5 and 3/4.4.6.2 and their Bases to maintain voltage-based steam generator tube repair criteria for the tube support plate elevations for future cycles of operation. The amendment replaces a 1.0 volt repair limit which had been approved on an interim basis by License Amendment No. 184 (issued February 3, 1995) with a 2.0 volt repair limit. The amendment also includes additional changes to reflect the guidance provided in NRC Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.''

Date of issuance: April 1, 1996.

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

Amendment No: 198.

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

Date of initial notice in Federal Register: January 3, 1996 (61 FR 178) The January 4, March 1, March 5, March 7, March 11, March 27, and March 29, 1996, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the January 3, 1996 notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 14, 1995.

Brief description of amendment: The amendment consists of several changes to the instrumentation sections of the Clinton Power Station Technical Specifications. These changes were required due to engineering reanalyses or plant modifications. The affected instrumentation includes: (1) steam line flow high channels for the reactor core isolation cooling (RCIC) system, (2) ambient temperature channels in the residual heat removal (RHR) system heat exchanger rooms, (3) reactor vessel pressure channels that provide a permissive for operation of the shutdown cooling mode of the RHR system, and (4) RCIC storage tank water level instrument channels.

Date of issuance: April 10, 1996.

Effective date: April 10, 1996.

Amendment No.: 104.

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1631) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1996.

No significant hazards consideration comments received: No.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 15, 1995, as supplemented November 14, and December 20, 1995.

Brief description of amendment: The amendment modifies the Monticello Technical Specifications (TS) to: (1) revise the main steam line isolation valve leak rate test acceptance criterion to be based upon the combined maximum flow path leakage for all four main steam lines of 46 standard cubic feet per hour (scfh) in lieu of the current limit of 11.5 scfh per valve; (2) revise the operability test interval for the drywell spray header and nozzles from 5 years to 10 years; and (3) revise TS 3/4.7.a.2, Primary Containment Integrity, to remove information specific to the primary containment leakage rate testing program and adopt the requirements of 10 CFR Part 50, Appendix J, Option B, for Type A testing, while remaining under Appendix J, Option A, for Type B and C testing.

Date of issuance: April 3, 1996.

Effective date: April 3, 1996.

Amendment No.: 95.

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1632) The December 20, 1995, letter provided clarifying information that was within the scope of the initial notice and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 3, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 1, 1996 (supersedes December 11, 1995, application).

Brief description of amendment: The amendment modifies Technical Specification Section 4.7, Surveillance Requirements for Primary Containment Automatic Isolation Valves, by revising Surveillance Requirement 4.7.D.4 to require that the seat seals of the drywell and suppression chamber purge and vent valves be replaced every six operating cycles.

Date of issuance: April 9, 1996.

Effective date: April 9, 1996.

Amendment No.: 96.

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 8, 1996 (61 FR 9504). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 9, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 22, 1995.

Brief description of amendments: The amendments change Technical Specification 3.6.1.8, ''Drywell and Suppression Chamber Purge System,'' increasing the drywell and suppression chamber purge system operating time limit from 90 hours each 365 days to 180 hours each 365 days.

Date of issuance: March 29, 1996.

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

Amendment Nos.: 115 and 77.

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7555).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 29, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 9, 1996, as supplemented March 20, 1996.

Brief description of amendment: The proposed amendment would revise the Technical Specifications (TSs) to use an installed retractable overhead door assembly and change TS 3.9.3 to satisfy closure requirements for the containment equipment hatch during core alterations or fuel movement in the containment building. The retractable door is to be used as a functionally equivalent closure plate currently required by TS 3.9.3.

Date of issuance: April 1, 1996.

Effective date: April 1, 1996.

Amendment No.: 62.

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7557). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 1996.

No significant hazards consideration comments received: No.

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

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: August 18, 1995, as supplemented on November 1, 1995, February 14, March 14 (there are two supplemental letters with this date), and March 25, 1996.

Brief description of amendment: The amendment revises the Operating License (OL) to increase the authorized core power level from 2775 Megawatts thermal (MWt) to 2900 MWt. The amendment also approves changes to the technical specifications (TS) to implement uprated power operation.

Date of issuance: April 12, 1996.

Effective date: April 12, 1996.

Amendment No.: 133.

Facility Operating License No. NPF–12: Amendment revises the OL and TS.

Date of initial notice in Federal Register: December 6, 1995 (60 FR 62495). The original Federal Register notice included information from the licensee's November 1, 1995 supplemental letter. The February 14, March 14, and March 25, 1996 supplemental letters provided clarification and amplification of the analysis in the November 1, 1995 letter and were not outside the scope of the initial Federal Register notice. The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated April 12, 1996 and in a Safety Evaluation dated April 12, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

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

Date of application for amendments: December 30, 1992, as supplemented by letters dated September 7, 1993, August 17, 1994, and March 7, 1996.

Brief description of amendments: These amendments add a new technical specification (TS) 3/4.7.3.1, ''Component Cooling Water (CCW) Safety Related Makeup System,'' and its associated Bases. The new TS will ensure that sufficient CCW capacity is available for continued operation of safety-related equipment during normal conditions and design-basis events.

Date of issuance: April 11, 1996.

Effective date: April 11, 1996.

Amendment Nos.: Unit 2–129; Unit 3–118.

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

Date of initial notice in Federal Register: March 3, 1993 (58 FR 12268). The September 7, 1993, August 17, 1994, and March 7, 1996, letters provided additional clarifying information and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 11, 1996.

No significant hazards consideration comments received: No.

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

Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of application for amendment: December 12, 1995, and supplemented March 4, 1996 (TS 95–23).

Brief description of amendment: The amendment revises the TS surveillance requirements and bases to incorporate alternate S/G tube plugging criteria at tube support plate (TSP) intersections. The approach taken is based on guidance given in Generic Letter (GL) 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.'' The amendment is applicable for Cycle 8 operation only.

Date of issuance: April 3, 1996.

Effective date: April 3, 1996.

Amendment No.: 211.

Facility Operating License Nos. DPR–77: Amendment revises the technical specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 183) The March 6, 1996 supplemental letter provided clarifying information which did not change the proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 3, 1996.

No significant hazards consideration comments received: None

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

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

Date of application for amendment: February 27, 1996, as supplemented by letter dated March 1, 1996.

Brief description of amendment: The amendment allows the drywell personnel air lock shield doors to be open during Operational Conditions 1, 2, and 3 until the end of Operating Cycle 6.

Date of issuance: March 22, 1996.
Effective date: March 22, 1996.
Amendment No.: 84.
Facility Operating License No. NPF–58: This amendment approved a change to the design basis as described in the Updated Safety Analysis Report. Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 8982 dated March 8, 1996). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing BiWeekly Notice by March 18, 1996, corrected to April 5, 1996 (61 FR 10600 dated March 14, 1996), but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The March 1, 1996, supplemental letter provided additional clarifying information and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment and final no significant hazards consideration determination is contained in a Safety Evaluation dated March 22, 1996.

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

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

Date of amendment requests: November 21, 1995 (TXX–95288) as supplemented by letters dated December 15, 1995 (TXX–95306), and February 2, 1996 (TXX–96040).

Brief description of amendments: The amendments revised the core safety limit curves and revised N–16 Overtemperature reactor trip setpoints as a result of the reload analyses for CPSES Unit 2, Cycle 3. In addition, the minimum required Reactor Coolant System (RCS) flow was increased and an administrative enhancement was included in the footnotes of the RCS flow-low reactor trip function setpoint for both Units 1 and 2.

Date of issuance: April 1, 1996.
Effective date: April 1, 1996.
Amendment Nos.: Unit 1–49; Unit 2–35.
Facility Operating License Nos. NPF–87 and NPF–89. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 185) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 1, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 26, 1995.

Brief description of amendments: The amendments revise the Technical Specifications to increase the pressurizer safety valve lift setpoint tolerance and reduce the pressurizer high pressure reactor trip setpoint and allowable value.

Date of issuance: April 1, 1996.
Effective date: April 1, 1996.
Amendment Nos.: 200 and 181.
Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register*:* August 30, 1995 (60 FR 45189) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 1, 1996.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 8, 1996, as supplemented by letter dated March 26, 1996.

Brief description of amendment: This amendment reduces the calculated thermal design flow of the reactor coolant system and increases the trip setpoint of the low pressurizer pressure.

Date of issuance: April 4, 1996.
Effective date: April 4, 1996.
Amendment No.: 99.
Facility Operating License No. NPF–42: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 10389 dated March 13, 1996). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by April 12, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 4, 1996.

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

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 24, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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, and to the attorney for the licensee.

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

Arizona Public Service Company, et al., Docket No. STN 50–529, Palo Verde Nuclear Generating Station, Unit 2, Maricopa County, Arizona

Date of application for amendment: April 1, 1996, as supplemented by letter dated April 3, 1996.

Brief description of amendment: The amendment modifies Technical Specification (TS) 3/4.9.6 to temporarily allow the use of a hoist instead of the refueling machine for the movement of the fuel assembly at core location A–07.

Date of issuance: April 3, 1996.

Effective date: April 3, 1996.

Amendment No.: Unit 2—96.

Facility Operating License No. NPF–51: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 3, 1996.

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

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

NRC Project Director: William H. Bateman.

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

Date of application of amendments: April 2, 1996.

Brief description of amendments: The amendments revise Technical Specification (TS) Section 4.5.4, ''Penetration Room Ventilation System'' and TS Section 4.14, ''Reactor Building Purge Filters and Spent Fuel Pool Ventilation System.'' The change updates the industry guidance reference for testing charcoal absorber units for the system covered by those TS.

Date of Issuance: April 2, 1996.

Effective date: April 2, 1996, to be implemented within 30 days.

Amendment Nos.: 215, 215, and 212.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 2, 1996.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691.

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

NRC Project Director: Herbert N. Berkow.

Dated at Rockville, Maryland, this 17th day of April 1996.

For the Nuclear Regulatory Commission.

Steven A. Varga,

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

[FR Doc. 96–9925 Filed 4–23–96; 8:45 am]

BILLING CODE 7590–01–P

[Docket Nos. 50–245, 50–336 AND 50–423]

Millstone Nuclear Power Station; Establishment of Temporary Local Public Document Room

Notice is hereby given that the Nuclear Regulatory Commission (NRC) has designated the Waterford Public Library, Waterford, Connecticut, as a temporary local public document room (LPDR) for Northeast Nuclear Energy Company's Millstone Nuclear Power Station. The NRC's official full service LPDR, located at the Three Rivers Community Technical College, Thames Valley Campus, Norwich, Connecticut, is still open and operational.

Members of the public may now inspect and copy Millstone related documents dated April 1, 1996, forward at the Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385. The library is open on the following schedule: Monday through Thursday 9:00 a.m. to 9:00 p.m.; Friday 9:00 a.m. to 5:30 p.m.; and Saturday 9:00 a.m. to 5:00 p.m.

For further information, interested parties in the Waterford area may contact the LPDR directly through Mr. Vincent Juliano, Library Director, telephone number (860) 444–5805. Parties outside the service area of the LPDR may address their requests for records to the NRC's Public Document Room, Washington, DC 20555, telephone number (202) 634–3273.

Questions concerning the NRC's local public document room program or the availability of documents should be addressed to Ms. Jona Souder, LPDR Program Manager, Freedom of Information/Local Public Document Room Branch, Division of Freedom of Information and Publications Services, Office of Administration, U. S. Nuclear Regulatory Commission, Washington, DC 20555, telephone number (301) 415–7170 or toll-free 1–800–638–8081.

Dated at Rockville, Maryland, this 18th day of April, 1996.

For the Nuclear Regulatory Commission.

Carlton Kammerer,

Director, Division of Freedom of Information and Publications Services, Office of Administration.

[FR Doc. 96–10050 Filed 4–23–96; 8:45 am]

BILLING CODE 7590–01–P

Receipt of Petition for Director's Decision Under 10 CFR 2.206

Notice is hereby given that by letter dated March 5, 1996, Mr. C. Morris submitted a Petition pursuant to 10 CFR 2.206 requesting that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to all nuclear power plants.

The Petitioner requests that, within 90 days, the operating licenses of all nuclear power plants be suspended until such time as those licensees have discovered the reasons for the repeated errors in their electrical distribution system designs and in their undervoltage relay (UVR) set points, and provided convincing evidence that these deficiencies have been corrected. Since the Petitioner asserts that the situation is urgent, the request is being treated as one for immediate relief. The Petitioner also requests that the aforementioned evidence be submitted for review by a competent third party, and that if the NRC finds that licensees may safely operate with UVRs that do not remain properly set, it should do so in the context of a public meeting.

The Petition is being treated pursuant to 10 CFR 2.206 of the Commission's regulations and has been referred to the Director of the Office of Nuclear Reactor Regulation. As provided by Section 2.206, appropriate action will be taken