the performance assessment and regulatory oversight processes. These meetings are open to the public and all interested parties may attend and participate.

DATES: The workshop will be held from September 28 through October 1, 1998. The workshop will be held from 8:00 a.m. to 5:00 p.m. on September 28 through September 30, 1998. On October 1, 1998, the workshop will again start at 8:00 a.m., and is scheduled to conclude at 2:00 p.m.

ADDRESSES: The workshop will be held at the Bethesda Marriott, 5151 Pooks Hill Road, Bethesda MD 20814.

FOR FURTHER INFORMATION CONTACT: Timothy J. Frye at 301–415–1287 or David L. Gamberoni at 301–415–1144, Mail Stop: O–5H4, Inspection Program Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

SUPPLEMENTARY INFORMATION:

Background

In September 1997, the NRC began an integrated review of the assessment processes (IRAP) used for commercial nuclear power plant licensees. A cross-disciplinary team of NRC staff members was assembled to identify and evaluate potential improvements to the process used by the NRC to assess licensee performance. A process re-engineering approach was taken by the team to identify the desired objectives of a new assessment process, the attributes it should possess, and criteria to measure improvement over the existing assessment processes.

The team developed a conceptual design for a new integrated assessment process and presented it to the NRC Commissioners in Commission paper SECY–98–045, dated March 9, 1998. On April 2, 1998, the staff briefed the Commission on the concepts as discussed in the paper. On June 30, 1998, the Commission issued a staff requirements memorandum (SRM) in response to SECY–98–045, approving the staff's request to solicit public comment on the concepts presented in the Commission paper.

The NRC issued (1) background material on the concept developed for a new integrated assessment process and (2) other assessment tools such as trending methodology, financial indicators, and risk-informed assessment guidance in the report ''Concepts Developed by the Integrated Review of Assessment Process for Commercial Nuclear Power Plants,'' dated July 29, 1998. On August 7, 1998, the NRC issued a **Federal Register** Notice announcing a 60-day public comment period to solicit public comment on possible changes to the NRC's assessment and regulatory oversight processes.

In parallel with staff work on the IRAP and the development of other assessment tools, the industry has independently developed a proposal for a new assessment and regulatory oversight process. This proposal would take a risk-informed, performance-based approach to the inspection, assessment, and enforcement of licensee activities based on the results of a set of performance indicators. This proposal, which is being developed by the Nuclear Energy Institute, is further described in ''Minutes of the July 28, 1998, Meeting With the Nuclear Energy Institute to Discuss Performance Indicators and Performance Assessment,'' dated July 30, 1998.

Scope of the Public Workshops

The NRC will hold a four day workshop to develop improvements to the licensee performance assessment and regulatory oversight processes. As background information, concepts previously developed by the NRC and the industry for improving the performance assessment and regulatory oversight processes will be discussed. At the workshop, the NRC will present a framework that links various regulatory oversight activities, such as inspection and assessment, to the overall objective of the agency and the industry, which is to ensure the adequate protection of public health and safety.

Several fundamental issues will then be discussed in order to develop the attributes that a new process must meet. These fundamental issues will be discussed in focused breakout sessions and grouped by the following topics: (1) general policy issues involving safety performance expectations and regulatory oversight; (2) the use of risk insights in the assessment process; (3) the use of performance indicators and their integration with inspection results; and (4) the role of enforcement in regulatory oversight/range of NRC actions/communication of assessment results.

The attributes that result from the discussion of these fundamental issues will then be used by the workshop participants to develop the specific details for improvements to the performance assessment and regulatory oversight processes. This development activity will again occur in focused breakout sessions, with each group focused on the development of one aspect of the new process.

Workshop Pre-Registration

Attendees at the workshop are requested to pre-register with the NRC approximately three weeks before the workshop. In order to pre-register, please give your utility or group affiliation, list the names of the people planning to attend, and indicate which of the fundamental issue breakout sessions that members of your group are interested in participating. Attendees may pre-register in any of the following ways:

(1) Contact via e-Mail either Timothy J. Frye at TJF@NRC.GOV, or David L. Gamberoni at DLG2@NRC.GOV, or

(2) Submit the pre-registration information to either Timothy J. Frye or David L. Gamberoni via fax at 301–415–3707, or

(3) Send written pre-registration data to: U.S. Nuclear Regulatory Commission, Attn: Timothy J. Frye, Office of Nuclear Reactor Regulation, Mail Stop O–5H4, Washington, DC 20555–0001.

A block of hotel rooms has been reserved at the Bethesda Marriott for the use of the workshop participants. These rooms will be available until September 9, 1998, and should be reserved by contacting the hotel at 301–897–9400 and requesting a room in the ''NRC'' block. After September 9, 1998, reservations will be accepted on a space available basis.

Dated at Rockville, Maryland, this 20th day of August 1998.

For the Nuclear Regulatory Commission.

Frank P. Gillespie,

Director, Division of Inspection & Support Programs, Office of Nuclear Reactor Regulation.

[FR Doc. 98–22906 Filed 8–25–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 3, 1998, through August 14, 1998. The last biweekly notice was published on August 12, 1998 (63 FR 43200).

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

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

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

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

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

By September 25, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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 amendments request: July 20, 1998.

Description of amendments request: The amendment incorporates the changes described below into the Technical Specifications (TS) for Calvert Cliffs Unit 2. Currently, Calvert Cliffs has four emergency diesel generators (EDGs), two per Unit, to provide the onsite emergency power supply for both Units. The Unit 2 EDGs rely on the Service Water (SRW) System to provide their cooling water. During the Unit 2 1999 Refueling Outage, Baltimore Gas and Electric Company will replace the SRW heat exchangers on Unit 2. During the period of the replacement, no SRW cooling will be available for Unit 2. Therefore, both Unit 2 EDGs would be inoperable during the replacement work. Unit 1 will continue at full power operation during the Unit 2 refueling outage.

The loss of both EDGs on Unit 2 presents several challenges. First, a number of outage activities require an EDG to be operable. BGE proposes to provide an alternate cooling water supply to maintain the EDGs operable to fulfill the TS requirements. One EDG will be provided with cooling water from the Unit 1 SRW System. The other EDG will be provided with cooling water from an independent external cooling system. Second, Unit 1 is scheduled to be in Mode 1 operation during this time. The No. 12 Control Room Emergency Ventilation System, No. 12 Control Room Emergency Temperature System, and a Hydrogen Analyzer are affected by this work because they obtain their emergency power from a Unit 2 EDG. These components support Unit 1 continued operation. Therefore, the loss of both Unit 2 EDGs would impact operations on both units.

There are several issues associated with this change that create an Unreviewed Safety Question (USQ) as defined by 10 CFR 50.59. There is an increase in the probability of a malfunction due to the use of an independent cooling system that is non-safety-related and unprotected from seismic or tornado events. The reliance of a Unit 2 EDG on Unit 1 SRW results in the increase of the probability of a malfunction, also. Additionally, these SRW lineups affect the probability of a malfunction for other equipment that relies on SRW during an outage. The approval of these USQs, will permit a TS Bases change to the description of an operable EDG while Unit 2 is in Modes 5 and 6.

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

The EDGs are used to mitigate the consequences of an accident. They are designed to start and load safety-related loads within a specified time period. There are two EDGs for Unit 2. Only one is required during the refueling outage, since a single failure criterion does not apply during this time. However, it is desirable for defense-in-depth and shutdown safety reasons to keep both EDGs operable. Additionally, one of the EDGs supports operable equipment on Unit 1 that remains at power. We are proposing an amendment that would allow the EDGs to continue to be operable with an alternate cooling water supply. Other than the change in cooling water supply, we are not affecting or modifying the operation of the EDGs. The EDGs are not an accident initiator for any previously evaluated accident. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

The EDGs are designed to mitigate the consequences of an accident. They will continue to perform that function while being supplied with an alternate source of cooling water. The consequences of a design basis accident during the period when the alternate cooling water is being supplied is not increased because the operation of the EDGs has not been adversely affected. Any additional electrical loads (such as cooling tower pumps and fans) or additional cooling loads (such as additional SRW flow to the No. 2A EDG) have been evaluated and found to be acceptable under conditions postulated to exist during the outage. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The EDGs are not being modified by this proposed change nor will any unusual operator actions be required. The EDGs will continue to operate in the same manner as before. However, the cooling water supplies have been altered and were evaluated under the provisions of 10 CFR 50.59 and determined to result in a USQ. These USQs are evaluated below.

The first identified USQ is due to the realignment of a Unit 1 SRW subsystem to also support a Unit 2 EDG (2A). This alignment will rely on two control valves (one to each EDG) to function properly in order to provide adequate SRW flow to both EDGs. If one of the valves should fail open, it may result in insufficient SRW flow or increased SRW temperatures, as the EDGs share the same cooling supply. This is an increase in the probability of a malfunction because the operability of a EDG relies on both control valves performing properly. We believe that this is an acceptable condition because the control valves and their air supply are safety-related and will be performing their design function. The control valves are not being modified by the temporary configuration nor will any operator action be required. The control valves will continue to operate in the same manner. Therefore, because the malfunction is the same as previously identified for these valves and only the probability has increased, a new or different type of accident has not been created.

The next USQ identifies a condition where a Unit 2 EDG is dependent on a Unit 1 EDG for cooling water. The Unit 1 EDG powers the pump for the cooling water system that will now provide cooling to both EDGs. Although the consequences of a loss of cooling water is the same (i.e., the EDG fails), the probability of a malfunction for the Unit 2 EDG has increased because it now depends on the Unit 1 EDG to maintain its operability. We believe that this is an acceptable condition because the Unit 1 EDG is safety-related and is proven reliable through testing.

Additionally, the EDG will not be operated in a manner different than it is currently. It is not being modified by the proposed change nor will any additional operator actions be required. A failure analysis shows that failure of the No. 1B EDG will not result in the total loss of any safety function for either unit. Therefore, the possibility of a new or different type of accident has not been created.

A USQ has been identified related to the use of a temporary cooling system to provide cooling to an EDG. The cooling system what is proposed is not safety-related and is not protected from natural phenomenon. This leads to an increase in the probability of a malfunction because the cooling system is more likely to fail than a safety-related, protected system. We believe that this is an acceptable condition for the limited time we propose to use the cooling system. The consequences of a cooling system failure are no different than those of a failure of the SRW System. The events most likely to cause the cooling system to fail are seismic events and severe weather. Severe weather is not highly probable during this time of year. Significant seismic events are not probable on this part of the east cost. The cooling tower has been used before at Calvert Cliffs to support testing of the EDGs during outages. The cooling tower will have enhanced design features that will improve its reliability, such as two pumps. The piping provided to and from the cooling system will be steel and will be provided with flexible joints making it rugged and flexible. Additionally, the cooling tower will be placed close to the Auxiliary Building and the makeup water piping will be run underground for part of its length. These measures help to protect the cooling tower and its piping from severe weather events. The EDG is not being altered by this temporary configuration. It will continue to operate as before. No additional operator action is required for the cooling tower to perform its function. Therefore, the possibility of a new or different type of accident has not been created.

This USQ exists because the piping from the cooling tower to the EDG is not safety-related and could break, causing a flood in the EDG room. This creates an increase in the probability of a malfunction because of the increased probability of flooding in the room. We believe that this increase is acceptable because the piping is constructed from rugged materials and is flexibly connected to the EDG. This reduces the chance that flooding will occur. If flooding were to occur and the contents of the cooling system were spilled into the room, it would not impact safety-related components in the room because the water would not be deep enough. Therefore, the possibility of a new or different accident has not been created.

Therefore, the possibility of a new or different type of accident from any accident previously evaluated has not been created.

3. Would not involve a significant reduction in a margin of safety. The operability of the EDGs in Modes 5 and 6 ensures that emergency power is available to mitigate the consequences of a fuel handling accident and a boron dilution accident. Additionally, it provides emergency power for shutdown cooling and spent fuel pool cooling. One of the Unit 2 EDGs provides power to the shared Control Room Emergency Ventilation System, Control Room Emergency Temperature System, and the Hydrogen Analyzer needed to Support Unit 1 power operation. The proposed changes do not affect the function of the EDGs. Because of the increased probability of a malfunction of equipment important to safety (SRW support for the EDGs), the margin of safety is reduced. However, the reduction is not significant. As described above, each USQ has been evaluated and determined to not have a significant impact on safety.

To provide additional assurance that all reasonable steps have been taken to ensure the operability of the Unit 2 EDGs while in the temporary configuration, the following actions will be taken in addition to the installation of the temporary modifications as described above:

To prevent the loss of the normal power supply to the Control Room Emergency Ventilation System and Control Room Emergency Temperature System, we will restrict maintenance activities on three of the four offsite transmission lines until the Unit 2 EDGs are returned to normal configuration.

To monitor risk, Unit 1 and 2 equipment taken out-of-service during this period will be evaluated in the Unit 1 weekly quarterly system schedule evaluations.

To ensure that weather-related events cannot cause a loss of all emergency power on Unit 2 during periods of reduced inventory, the No. 2A EDG will remain operable during reduced inventory periods.

To ensure that backup power is available to any of the safety-related buses, the No. 0C Diesel Generator will not be taken out-of service for planned maintenance and will remain available to be connected to any of the safety-related buses.

We believe that the reduction in the margin of safety represented by this temporary license amendment is not significant based on our evaluation and management of plant risk, the reliability of the EDGs, the availability of redundant EDGs, the availability of the Station Blackout Diesel Generator and the mitigating features described above. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: S. Singh Bajwa, Director.

Duke Energy Corporation (DEC or licensee), Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: May 27, 1997, as supplemented by letters dated March 9, March 20, April 20, June 3, June 24, July 7, July 21, and July 22, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) of each unit to conform with NUREG–1431, Revision 1, ''Standard Technical Specifications—Westinghouse Plants.'' The Commission had previously issued a Notice of Consideration of Issuance of Amendments in the **Federal Register** on July 15, 1997 (62 FR 37940) covering all the proposed changes that were indeed within the scope of NUREG–1431. In DEC's May 27, 1997, submittal, there are proposed changes that are beyond the scope of NUREG–1431, which were, thus, not covered by the staff's July 15, 1997, notice. The following description and no significant hazard analysis covers a beyond-scope change.

The licensee proposed to change Section 3.4.6.1 regarding reactor coolant leakage detection systems; a system comprising diverse instruments such as gaseous radioactivity monitoring, containment floor and equipment sump monitoring, etc. In addition to the instruments specified by this section, the plant has other installed instruments such as monitors for humidity, temperature, etc., which can provide indication for reactor coolant leakage. Currently, this specification allows operation up to 30 days if the containment floor and equipment sump monitoring system is inoperable. The proposed change would impose a requirement to perform a precision water balance of the reactor coolant system every 24 hours during this period. The proposed change would also reduce the number of monitors required operable provided compensatory measures are performed or diverse instruments continue to be available.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analyses of the issue of no significant hazards consideration for each of the above proposed changes. The NRC staff has reviewed the licensee's analyses against the standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below.

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

evaluated? The proposed change will not affect the safety function of the subject systems. There will be no direct effect on the design or operation of any plant structures, systems, or components. No previously analyzed accidents were initiated by the functions of these systems, and the systems were not factors in the consequences of previously analyzed accidents. Therefore, the proposed change will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

The proposed change would not lead to any hardware or operating procedure change. Therefore, no new equipment failure modes or accidents from those previously evaluated will be created.

3. Will the change involve a significant reduction in a margin of safety? Margin of safety is associated with confidence in the design and operation of the plant. The proposed change to the TS do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

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

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

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: July 21, 1998.

Description of amendment request: The proposed change to the Technical Specifications would: (1) modify Specification 6.2.2.2(a) to provide some flexibility to accommodate unexpected absence of on-duty shift crew members, (2) eliminate reference to the Manager, Plant Operations in Specification 6.2.2.2(j) as the position has been eliminated, (3) reduce the maximum time in which to forward audit reports to the responsible manager from 60 days to 30 days, (4) replace the term ''Vice President'' with the term ''Corporate Officer'' in several places in Section 6, and (5) correct several typographical errors.

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.

The activity does not alter the design, function or manner of operation of any structures, systems or components. Therefore, this activity does not increase the probability 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 a different type than any previously identified in the SAR.

The activity does not alter the design, function, or manner of operation of any structures, systems or components. Therefore, this 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 is not reduced.

The activity does not alter the design, function or manner of operation of any structures, systems or components. In addition, a decrease in staff for a short period of time on limited occasions is not safety significant and permitted by 10 CFR 50.54 (m). Therefore, this activity will not reduce 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: 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: Cecil O. Thomas.

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

Date of amendment request: July 23, 1998.

Description of amendment request: Amend facility license to establish that the existing Safety Limit Minimum Critical Power Ratio (SLMCPR) contained in Technical Specification 2.1.A is applicable for the next operating cycle (Cycle 17).

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

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

The derivation of the Cycle 17 SLMCPR for Oyster Creek for incorporation into the TS, and its use to determine cycle-specific thermal limits, has been performed using NRC-approved methods. Additionally, interim implementing procedures, which incorporate cycle-specific parameters, have been used. Based on the use of these calculations, the Cycle 17 SLMCPR of 1.09 will not increase the probability or consequences of an accident.

The basis of the MCPR Safety Limit calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. A SLMCPR of 1.09 preserves adequate margin to transition boiling and fuel damage in the event of a postulated accident. The probability of fuel damage is not increased.

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

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

The MCPR Safety Limit is a Technical Specification numerical value designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. The limit cannot create the possibility of any new type of accident. The Cycle 17 SLMCPR has been calculated using NRC-approved methods. Additionally, interim procedures, which incorporate cycle-specific parameters, have been used. Therefore, the proposed TS change does not create the possibiliy of a new or different kind of accident, from any accident previously evaluated.

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

The margin of safety as defined in the TS Bases will remain the same. The Cycle 17 SLMCPR is calculated using NRC-approved methods, which are in accordance with the current fuel design and licensing criteria. Additionally, interim implementing procedures, which incorporate cycle-specific parameters, have been used. The MCPR Safety Limit remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving fuel cladding integrity. Therefore, the proposed TS change does not involve a reduction in a margin of safety.

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

Local Public Document Room location: 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: Cecil O. Thomas.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: August 6, 1998.

Description of amendment request: The proposed amendment would allow changes to the Updated Safety Analysis Report (USAR) to reflect the as-built configuration of the reactor building isolation dampers. These changes would clarify the USAR discussion of secondary containment isolation and revise the calculated offsite dose consequences resulting from a postulated refueling accident. No changes to the Technical Specifications (TS) are required; the TS Bases, § 3.6.4.2, will be revised under the licensee's Bases control program to reflect the changes in the USAR analysis.

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

The enclosed proposed license amendment for the as-built design of the Secondary Containment (Reactor Building) isolation dampers is judged to involve no significant hazards based on the following:

1. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The existing plant design does not involve a significant increase in the probability of an accident previously evaluated in the Updated Safety Analysis Report (USAR). The current configuration does not affect the performance and reliability of the Secondary Containment and the Reactor Building Isolation and Control System or any system interface in a way that could lead to an accident occurring. The current configuration and analysis do not affect any accident precursors or initiators, and therefore, does not increase the probability of an accident.

The present plant configuration also does not involve a significant increase in the consequences of an accident previously evaluated in the USAR. The current design will require a clarification to the Secondary Containment safety design basis as described in the USAR to reflect the as-built configuration and analysis of the plant by stating that the Reactor Building Isolation and Control System is designed to limit the release of fission products through the normal ventilation discharge path during a postulated Refueling Accident.

The original analysis determined that the consequences of the Refueling Accident were significantly less than 1 Rem to the thyroid and whole body (maximum off-site dose). When this analysis was revised to account for the 90 second motor-operated damper closure time, the calculated whole body off-site dose increased, but was still less than 1 Rem; the calculated off-site dose to the thyroid, however, increased to 2.7 Rem. While this change in the analysis represents an order of magnitude increase in consequences (thyroid dose increase from 17 milliRem to 2.7 Rem), the actual increase is minimal because this increase in consequences is still less than 1 percent (1%) of the limits specified in 10 CFR 100. Thus the consequences still remain well within the regulatory threshold specified in 10 CFR 100 and thus pose no undue hazard to the health and safety of the public. This proposed amendment does not alter the Control Room dose from that which was submitted to the NRC in support of Amendment 167.

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

This proposed license amendment is administrative in nature in that it reflects the effects of a revised analysis for the Refueling Accident, which is an accident previously analyzed as a Design Basis Accident (DBA) in the SAR, based on the present configuration of the plant. The current configuration does not create the possibility of a new or different kind of accident from any accident previously evaluated in the USAR. The proposed license amendment does not introduce any new equipment or hardware changes, nor does it require existing equipment or systems to perform a different type of function than they are presently designed to perform. The as-built configuration does not introduce any new mode of plant operation, thus there are no new accident failure paths created.

The as-built configuration does not affect any accident precursors or initiators and does not create the possibility of a new or different kind of accident.

3. Does not create a significant reduction in the margin of safety.

The present plant configuration does not involve a significant reduction in a margin of safety. Technical Specification Bases section 3.2.D.2, Reactor Building Isolation and Standby Gas Treatment (SGT) Initiation, states that the trip settings for the Reactor Building exhaust plenum radiation monitors are based on initiating normal ventilation system isolation and SGT System operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path, but rather all the activity is processed by the SGT System. This basis statement remains true unless there is a single failure of the air-operated Secondary Containment isolation damper. Under single failure conditions there would be the potential for a limited release through the normal ventilation system prior to complete isolation of the secondary containment and initiation of the SGT System.

The significance of this change is minimal, as Technical Specification requirements to isolate Secondary Containment are still met. The overall function of the Secondary Containment and Reactor Building Isolation and Control System, in conjunction with other accident mitigation systems, is to limit fission product release during and following postulated DBAs. High radiation in the Secondary Containment exhaust is an indication of possible gross failure of the fuel cladding, possibly due to a Refueling Accident. The trip settings for the Reactor Building (Secondary Containment) radiation monitors are such that initiation of secondary containment isolation and SGT would still occur in sufficient time (within 90 seconds of detection) to maintain postulated off-site releases well within the limits of 10 CFR 100. As stated previously, the effects of the 90 second motor-operated damper closure time on Control Room dose have already been taken into consideration in the District's submittals supporting Amendment 167.

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: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305.

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Project Director: John N. Hannon.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: August 4, 1998.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) relating to the Condensate Storage Tank (CST) and also add a new TS section that would establish requirements for the atmospheric steam dump valves (ASDVs) to assure their operability. The applicable TS Bases section for the CST would also be changed to reflect the proposed changes and a new TS Bases section would be added to discuss the new TS section for the ASDVs.

Specifically, the proposed changes would modify TS 3.7.1.3, ''Plant

Systems—Condensate Storage Tank,'' by increasing the minimum required CST level from 150,000 gallons to 165,000 gallons to account for the discharge nozzle pipe elevation above the tank bottom and vortex formation in the CST at the auxiliary feedwater supply piping entrance. TS 3.7.1.7, ''Plant Systems—Atmospheric Steam Dump Valves,'' would be added to provide the requirements necessary to assure that the ASDVs will be available to either maintain the unit in hot standby or cool down the unit to shutdown cooling entry conditions if the condenser steam dump valves are not available. As previously noted, the TS Bases would be modified to reflect the 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:

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

The proposed change to increase the minimum required Condensate Storage Tank (CST) level of Technical Specification 3.7.1.3 will ensure sufficient water is available for the Auxiliary Feedwater (AFW) System to function as designed to mitigate design basis accidents. There will be no adverse effect on equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to add a Technical Specification for the Atmospheric Steam Dump Valves (ASDVs) will provide additional assurance that the ASDVs will be available to either maintain the unit in hot standby, or cool down the unit to Shutdown Cooling (SDC) entry conditions if the condenser steam dump valves are not available. The proposed change does not alter the way any structure, system, or component functions. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no adverse effect on any of the design basis accidents previously evaluated. Therefore, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to increase the minimum required CST level will ensure the AFW System will function as designed to mitigate design basis accidents. The proposed change to add a Technical Specification for the ASDVs will provide additional assurance that the ASDVs will be available, if needed. There will be no adverse effect on equipment important to safety. Therefore, there will be no significant reduction of margin of safety as defined in the Bases for Technical Specifications affected by these proposed changes.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Deputy Director: William M. Dean.

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: March 18, 1998.

Description of amendment request: The proposed amendment would revise the Bases for Technical Specification (TS) 3/4.6.2.1, ''Containment Spray System,'' of the combined technical specifications for the Diablo Canyon Power Plant, Unit Nos. 1 and 2, to clarify that containment spray is not required to be actuated during recirculation, but may be actuated at the discretion of the Technical Support Center. Additionally, the Bases would be clarified to state that the ability to spray containment using the residual heat removal (RHR) system is demonstrated by opening the RHR Spray Ring Cross Connect Valve 9003 A or B. The Bases will also be clarified to state that flow to the spray headers can be established with only one operable RHR pump by closing the cold leg discharge valve 8809 A or B.

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.

Containment spray (CS) in the recirculation mode of post-loss-of-coolant accident (LOCA) safety injection (SI) is used only after the accident has already occurred. Its availability or unavailability is unrelated to, and is not a precursor for, an accident that has already been initiated. The availability or unavailability of CS recirculation spray does not involve any physical change in plant systems, structures, or components, and there is no change in preaccident operating procedures, so there is no change to the probability of an accident occurring as a result of any such changes. The recirculation mode of emergency core cooling is only used following a LOCA; therefore, an evaluation of the effects of the use or absence of CS in the recirculation mode applies only to a LOCA and not to any other type of accident analyzed in the Final Safety Analysis Report (FSAR).

The peak post-LOCA containment pressure and temperature conditions occur prior to the recirculation phase of SI, and are not affected by CS operation during the recirculation mode of SI. The long term pressure and temperature profiles are slightly increased if recirculation spray is unavailable but are still within the dose analysis and equipment qualification requirements. There is no effect on the offsite dose analysis or on equipment operability.

If CS is not operated in the recirculation mode, there is no reduction in the amount of emergency core cooling system (ECCS) water pumped into the reactor vessel. Since the flow to the reactor is not reduced, core cooling is not adversely affected if recirculation spray is not used. If recirculation spray is used under Technical Support Center (TSC) direction with only one train of residual heat removal (RHR) in operation, ECCS flow to the reactor will be reduced, but analysis has shown that the flow to the reactor in this situation is still in excess of that needed to supply the required core cooling. Therefore, although it is not required, it would still be possible to establish CS in the recirculation mode with only one train of RHR in operation, if considered desirable by the TSC.

From the above discussion, it can be seen that the consequences of an accident analyzed in the FSAR are not increased because the absence of recirculation spray has no effect on the dose analyses and the effect on other accident parameters is within limits.

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 possibility of a malfunction of a different type than previously evaluated is not created for the following reasons:

Data provided in the FSAR can be used to determine that the iodine removal function for the CS system is completed in approximately 26 minutes, and prior to completing switchover to the recirculation mode after a LOCA. The statements in previous revisions of the FSAR that recirculation spray will continue for 2 hours to remove iodine are considered to be descriptive in nature, explaining an additional capability of the CS system, but not relied upon or evaluated in the FSAR.

The post-LOCA containment environmental conditions without recirculation spray remain bounded by those for which safety-related equipment inside containment is qualified; therefore, there is no resulting increase in the probability that it will malfunction. There is no other new mechanism created by the unavailability of recirculation spray that would lead to any greater probability of malfunction of safety-related equipment.

The peak post-LOCA containment pressure and temperature conditions occur prior to the recirculation phase of SI, and are not affected by CS operation during the recirculation mode of SI. Also, the Diablo Canyon Power Plant (DCPP) design bases and accident analyses do not assume any contribution to post-accident containment hydrogen mixing from recirculation spray. The DCPP design basis has always assumed that hydrogen mixing is achieved by containment fan cooler unit operation alone.

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.

Technical Specification (TS) 3/4.6.2.1, ''Containment Spray System,'' requires the operability of two trains of CS with each train capable of taking suction from the refueling water storage tank (RWST) and transferring spray function to an RHR train taking suction from the containment sump. With the proposed changes, the capability to perform the required alignment remains unaffected. However, the ability to actually provide CS in the recirculation mode of SI is limited by procedure in the event of failure of a train of auxiliary saltwater, component cooling water, or RHR. This does not affect the margin of safety as defined in the TS Bases. The Bases for CS operability are to ensure pressure reduction, cooling capability, and iodine removal from the containment atmosphere consistent with the assumptions used in the safety analyses.

All pressure reduction, cooling, and iodine removal parameters assumed in the accident analyses continue to bound those resulting in the event that recirculation spray is not used. The accident analyses require that the peak post-accident pressure does not exceed 47 psig, and that post-accident pressure be reduced to less than half the peak within 24 hours. These requirements are still met, but the long term pressure is slightly higher. Since these requirements are based on minimizing leakage rates and on environmental qualification concerns, and since the leakage rate in the offsite dose analysis and pressures for which safety-related equipment inside containment is qualified still bound the analysis results, a slightly higher long term pressure has no effect on safety margins. Although the long term temperature profile increases slightly with no recirculation spray, the equipment is still environmentally qualified for these temperatures, so again margin is maintained. The use of recirculation spray is not credited in the offsite or control room dose analyses since the containment atmospheric iodine decontamination factor reaches 1000 prior to the time recirculation spray is placed in service, so there is no loss of margin in the offsite and control room dose analyses. None of the accident analysis limits are exceeded in the absence of recirculation spray.

The function of CS to inject NaOH into the containment atmosphere and sump is not affected by the proposed changes. The same amount of RWST water will be pumped into the containment via the CS system for a given size LOCA with or without recirculation spray, so the same amount of NaOH is injected into the containment, and hence there is no effect on sump pH, iodine retention, or the dose analysis.

In the event that recirculation spray is established under TSC direction with only one train of RHR in operation, there is no reduction in the margin of safety from the resulting reduced flow to the core since analysis has demonstrated that even with no RHR flow to the RCS, the resulting flow to the core will still be greater than that required to maintain adequate core cooling and maintain peak clad temperatures within limits.

The functions specified for the CS system in the TS Bases are to ensure post-accident pressure reduction, cooling capability, and iodine removal from the containment atmosphere consistent with the assumptions used in the safety analyses. Since these functions are maintained within the limits of the safety analyses even in the absence of recirculation spray, the operability of the CS system as required by TS 3.6.2.1 is maintained.

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 & Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: William H. Bateman.

Pennsylvania Power and Light Company, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: August 5, 1998.

Description of amendment request: The amendment to Unit 2 Technical Specifications (TS) involves the addition of a new section entitled ''Oscillation Power Range Monitoring (OPRM) Instrumentation'' and revisions to Section 3.4.1 ''Recirculation Loops Operating'' to remove the specifications related to thermal power stability which will not be required after the installation of the OPRM instrumentation. Unit 2 is currently operating under Interim Corrective Actions (ICAs) defined in TS 3.4.1 that specify restrictions on plant operation and actions by operators in response to instability events. The OPRM system provides an automatic long-term solution to the instability issue and eases the burden on the operator.

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.

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

The OPRM most directly affects the [Average Power Range Monitor] APRM and [Local Power Range Monitor] LPRM portions of the Power Range Neutron Monitoring system. Its installation does not affect the operation of these sub-systems. None of the accidents or equipment malfunctions affected by these sub-systems are affected by the presence or operation of the OPRM.

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux changes. The APRM Fixed Neutron Flux-High function is capable of generating a trip signal to prevent fuel damage or excessive reactor pressure. For the [American Society of Mechanical Engineers] ASME overpressurization protection analysis in [Final Safety Analysis Report] FSAR Chapter 5, the APRM Fixed Neutron Flux-High function is assumed to terminate the main steam isolation valve closure event. The high flux trip, along with the safety/relief valves, limit the peak reactor pressure vessel pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis in Chapter 15 takes credit for the APRM Fixed Neutron Flux-High function to terminate the CRDA. The Recirculation Flow Controller Failure event (pump runup) is also terminated by the high neutron flux trip. The APRM Fixed Neutron Flux-High function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the Safety Limits (e.g., [Minimum Critical Power Ratio] MCPR and Reactor pressure) being exceeded.

The installation of the OPRM equipment does not increase the consequences of a malfunction of equipment important to safety. The APRM and [Reactor Protection System] RPS systems are designed to fail in a tripped (fail safe) condition; the OPRM will have no affect on the consequence of the failure of either system. An inoperative trip signal is received by the RPS any time an APRM mode switch is moved to any position other than Operate, an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs. These functions are not specifically credited in the accident analysis, but are retained for the RPS as required by the NRC approved licensing basis.

The OPRM allows operation under current operating conditions presently restricted by the current Technical Specifications by providing automatic suppression functions in the area of concern in the event an instability occurs. The consequences of any accident or equipment malfunction are not increased by operating under those conditions. Although protected by the OPRM from thermal-hydraulic core instabilities above 30% core power, operation under natural core recirculation conditions is not allowed. No accidents or transients of a type not analyzed in the FSAR are created by operating under these conditions with the protection of the OPRM system.

This change does not increase the probability of an accident as previously evaluated. The OPRM is designed and installed to not degrade the existing APRM, LPRM, and RPS systems. These systems will still perform all of their intended functions. The new equipment is tested and installed to the same or more restrictive environmental and seismic envelopes as the existing systems.

The new equipment has been designed and tested to the electromagnetic interference (EMI) requirements of Reference 2, which assures correct operation of the existing equipment. The new system has been designed to single failure criteria and is electrically isolated from equipment of different electrical divisions and from non-1E equipment. The electrical loading is within the capability of the existing power sources and the heat loads are within the capability of existing cooling systems. The OPRM allows operation under operating conditions presently forbidden or restricted by the current Technical Specifications. No other transient or accident analysis assumes these operating restrictions.

Based upon the analysis presented above, PP&L concludes that the proposed action does not involve an 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.

This proposal does not create the probability of a new or different type of accident from any accident previously evaluated. The OPRM system is a monitoring and accident mitigation system that cannot create the possibility for an accident.

The OPRM will allow operation in conditions currently restricted by the current Technical Specifications. Although protected by the OPRM from thermal-hydraulic core instabilities above 30% core power, operation under natural circulation conditions is not allowed. No accidents or transients of a type not analyzed in the FSAR are created by operating under these conditions with the protection of the OPRM system. No new failure modes of either the new OPRM equipment or of the existing APRM equipment have been introduced. Quality software design, testing, implementation and module self-health testing provides assurance that no new equipment malfunctions due to software errors are created. The possibility of an accident of a new or different type than any evaluated previously is not created.

The new OPRM equipment is designed and installed to the same system requirements as the existing APRM equipment and is designed and tested to have no impact on the existing functions of the APRM system. Appropriate isolation is provided where new interconnections between redundant separation groups are formed. The OPRM modules have been designed and tested to assure that no new failure modes have been introduced.

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

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

There has been no reduction in the margin of safety as defined in the basis for the Technical Specifications. The OPRM system does not negatively impact the existing APRM system. As a result, the margins in the Technical Specifications for the APRM system are not impacted by this addition.

Current operation under the ICAs provides an acceptable margin of safety in the event of an instability event as the result of preventive actions and Technical Specification controlled response by the control room operators. The OPRM system provides an increase in the reliability of the protection of the margin of safety by providing automatic protection of the MCPR safety limit, while the protection burden is significantly reduced for the control room operators. This protection is demonstrated as described above, and in the NRC reviewed and approved Topical Reports NEDO–32465–A and CENPD–400–P–A.

Replacement of the ICA operating restrictions from Technical Specifications with the OPRM system does not affect the margin of safety associated with any other system or fuel design parameter.

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

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

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

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

NRC Project Director: Robert A. Capra.

Southern Nuclear Operating Company, Inc., 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 amendments request: July 22, 1998.

Description of amendments request: The proposed amendments would change Technical Specification Tables 3.3.6.1–1 and 3.3.6.2–1 by increasing the Allowable Values for the high radiation trip for the exhaust monitors for the reactor building and the refueling floor.

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

The Unit 1 and Unit 2 reactor building and refueling floor ventilation exhaust radiation monitors perform no function in preventing, or decreasing the probability of, a previously evaluated accident. The monitors are designed to monitor ventilation exhaust for indications of a release of radioactive material resulting from a design basis accident and initiate appropriate protective actions. Because the proposed changes affect only the ventilation exhaust radiation monitors, the probability of an accident previously evaluated remains the same.

The function of the reactor building and the refueling floor ventilation exhaust radiation monitors, in combination with other accident mitigation systems, is to limit fission product release during and following postulated design basis accidents. The proposed new Allowable Values for the high radiation trip will continue to ensure the offsite doses resulting from a design basis accident do not exceed the NRC-approved licensing basis and FSAR [Final Safety Analysis Report] limits. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes increase the radiation level at which the ventilation exhaust monitors actuate; however, the manner in which their actuation logic functions and the systems that isolate or actuate as a result are unaffected by the proposed changes. Furthermore, the ventilation exhaust monitors will continue to perform their design function of limiting offsite doses to NRC-approved licensing basis and FSAR limits at the higher Allowable Values. Therefore, the proposed changes cannot create the possibility of a new or different kind of accident from any previously evaluated.

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

The Bases for Unit 1 and Unit 2 Technical Specification Tables 3.3.6.1–1 and 3.3.6.2–1 state that the Allowable Values for the reactor building and refueling floor ventilation exhaust radiation monitors ''are chosen to ensure radioactive releases do not exceed offsite dose limits.'' The proposed Allowable Values ensure the radiation monitors actuate at a radiation level sufficient to ensure offsite doses are within the NRC-approved licensing basis and FSAR limits. The proposed Allowable Values comply with the margin of safety defined in the Technical Specifications Bases for the ventilation exhaust radiation monitors; therefore, the proposed changes do not reduce a margin of safety.

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

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

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

NRC Project Director: Herbert N. Berkow.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: July 7, 1998.

Description of amendment request: The proposed amendment would revise the spent fuel pool criticality analysis and rack utilization schemes by allowing credit for spent fuel pool soluble boron.

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 presence of soluble boron in the spent fuel pool water for criticality control does not increase the probability of a fuel assembly drop accident in the spent fuel pool. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water.

The criticality analysis shows the consequences of a fuel assembly drop accident in the spent fuel pool are not affected when considering the presence of soluble boron. The rack K_{eff} [K effective] remains less than or equal to 0.95.

There is no increase in the probability of an accident. The proposed change does allow a greater number of fuel storage configurations in the spent fuel pool. While this could increase the probability of a fuel misloading, the presence of sufficient soluble boron in the spent fuel pool precludes criticality as a result of the misloading. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification spent fuel rack storage configuration limitations.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks. The criticality analyses demonstrate that the pool K_{eff} will remain less than or equal to 0.95 following an accidental misloading due to the boron concentration of the pool. The proposed Technical Specification limitation will ensure that an adequate spent fuel pool boron concentration is maintained.

There is no increase in the probability of the loss of normal cooling to the spent fuel pool water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the spent fuel pool water.

Reactivity changes due to spent fuel pool temperature changes have been evaluated. The basic case criticality analysis covers a ''normal'' spent fuel pool temperature range of 50 degrees F to 160 degrees F. Spent fuel pool temperature accidents are considered outside the normal temperature range extending from 32 degrees F to 240 degrees F. In all spent fuel pool temperature accident cases, sufficient reactivity margin is available to the 0.95 K_{eff} limit without requiring additional soluble boron above the base case level. Because adequate soluble boron will be maintained in the spent fuel pool water to maintain K_{eff} less than or equal to 0.95, the consequences of a loss of normal cooling to the spent fuel pool will not be increased.

Therefore, based on the conclusions of the above analysis, 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.

Spent fuel handling accidents are not new or different types of accidents, they have been analyzed in Section 15.7.4 of the Updated Final Safety Analysis Report.

Criticality accidents in the spent fuel pool are not new or different types of accidents, they have been analyzed in the Updated Final Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments for fuel enrichments up to and exceeding the nominal 4.95 weight percent U^{235} [Uranium-235] that is assumed for the proposed change.

Current Technical Specifications contain limitations on the spent fuel pool boron concentration. The actual boron concentration in the spent fuel pool has been maintained at a higher value. The proposed changes to the Technical Specifications establish new boron concentration requirements for the spent fuel pool water consistent with the results of the new criticality analysis (Attachment 2).

Since soluble boron has always been maintained in the spent fuel pool water, and is currently required by Technical Specifications, the implementation of this new requirement will have little effect on normal pool operations and maintenance. A dilution of the spent fuel pool soluble boron has always been a possibility; however, it was shown in the spent fuel pool dilution evaluation (Attachment 3) that there are no credible dilution events for which the spent fuel pool K_{eff} could increase to greater than 0.95. Therefore, the implementation of new limitations on the spent fuel pool boron concentration will not result in the possibility of a new kind of accident.

The proposed changes to Technical Specifications 3.9.13, 4.9.13, and 5.6 continue to specify the requirements for the spent fuel rack storage configurations. Since the proposed spent fuel pool storage configuration limitations will be similar to the current ones, the new limitations will not have any significant effect on normal spent fuel pool operations and maintenance and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the spent fuel pool loading configuration meets specified requirements.

The misloading of a fuel assembly in the required storage configuration has been evaluated. In all cases, the rack K_{eff} remains less than or equal to 0.95. Removal of an [sic] Rod Control Cluster Assembly from a checkboard storage configuration has been analyzed and found to be bounded by the misloading of a fuel assembly.

As discussed above, the proposed changes will not create the possibility of a new or different kind of accident. There is no significant change in plant configuration, equipment design or equipment.

Under the proposed amendment, no changes are being made to the racks themselves, any other systems, or to the physical structures of the Fuel Handling

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

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

The Technical Specification changes proposed by this License Amendment Request and the resulting spent fuel storage operation limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis (Attachment 2) performed in accordance with Westinghouse spent fuel rack criticality analysis methodology.

While the criticality analysis utilized credit for soluble boron, storage configurations have been defined using 95/95 K_{eff} calculations to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions and to provide subcritical margin such that the spent fuel pool K_{eff} is maintained less than or equal to 0.95.

The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to K_{eff} exceeding 0.95 has been evaluated (Attachment 3) and shown to be not credible. A safety evaluation has been performed which shows that dilution of the spent fuel pool boron concentration from 2500 ppm to 700 ppm is not credible. Also, the spent fuel rack K_{eff} will remain less than 1.0 (with a 95/95 confidence level) with the spent fuel pool flooded with unborated water. These safety analyses demonstrate a level of safety comparable to the conservative criticality analysis methodology required by Westinghouse WCAP–14416.

Based on the above evaluation, the South Texas Project concludes that the proposed changes to the Technical Specifications involve no significant hazards consideration.

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

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

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

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

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

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

Duke Energy Corporation, Docket No. 50–287, Oconee Nuclear Station, Unit No. 3, Oconee County, South Carolina

Date of amendment request: July 20, 1998.

Description of amendment request: The proposed amendment would extend, on a one-time basis, Technical Specification Surveillance 4.18.3 for hydraulic and mechanical snubber testing. The tests are required to be performed at a frequency of 18 months, with a maximum allowed frequency of 22 months, 15 days. The amendment would extend this to a maximum of 25 months.

Date of publication of individual notice in **Federal Register**: July 27, 1998 (63 FR 40137).

Expiration date of individual notice: August 26, 1998.

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

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.

CBS Corporation, Docket No. 50–22, Westinghouse Test Reactor, Waltz Mill, Pennsylvania

Date of application for amendment: December 22, 1997 supplemented on June 15, 1998.

Brief description of amendment: This amendment changes the legal name of the licensee for the Westinghouse Test Reactor from Westinghouse Electric Corporation to CBS Corporation.

Date of issuance: July 31, 1998.

Effective Date: July 31, 1998.

Amendment No: 7.

Facility Operating License No. TR–2: This amendment changes the legal name of the licensee for the Westinghouse Test Reactor from Westinghouse Electric Corporation to CBS Corporation.

Date of initial notice in **Federal Register**: July 15, 1998 (63 FR 38207).

The Commission has issued a Safety Evaluation for this amendment dated July 31, 1998.

No significant hazards consideration comments received: No.

Local Public Document: N/A.

Commonwealth Edison Company, Docket No. 50–249, Dresden Nuclear Power Station, Unit 3, Grundy County, Illinois

Date of application for amendment: May 6, 1998.

Brief description of amendment: The proposed amendment would amend Technical Specification (TS) 4.6.E to allow a one-time extension of the 40-month surveillance interval requirement to set pressure test or replace all Main Steam Safety Valves (MSSVs) to a maximum interval of 60 months as currently allowed by the American Society of Mechanical Engineers

(ASME) Boiler and Pressure Vessel Code (Code).

Date of issuance: August 7, 1998.

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

Amendment No.: 163.

Facility Operating License No. DPR–25: The amendment revised the TSs.

Date of initial notice in **Federal Register**: June 3, 1998 (63 FR 30263).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 1998.

No significant hazards consideration comments received: No.

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

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: November 2, 1994, as supplemented January 4, 1995, February 19, 1998, April 28, 1998, and June 5, 1998.

Brief description of amendment: The amendment revises the Technical Specifications that have become unnecessary due to previous approved amendments, make editorial changes, change managerial titles, update references and reporting requirements.

Date of issuance: August 12, 1998.

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

Amendment No.: 198.

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

Date of initial notice in **Federal Register**: November 8, 1995 (60 FR 56365).

The January 4, 1995, February 19, 1998, April 28, 1998, and June 5, 1998, letters provided clarifying information that 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 August 12, 1998.

No significant hazards consideration comments received: No.

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

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: January 28, 1998 (NRC–98–0011) as supplemented March 12 and June 9, 1998.

Brief description of amendment: The amendment revises Technical Specification 3.4.2.1, ''Safety/Relief Valves,'' changing the safety relief valve (SRV) setpoint tolerance from plus or minus 1 percent to plus or minus 3 percent. An associated footnote is revised to indicate that, although the as-found setpoint tolerance is now plus or minus 3 percent, the as-left settings of the SRVs shall be within plus or minus 1 percent of the specified setpoints prior to installation of the SRVs after testing. Bases section 3/4.4.2 is also revised.

Date of issuance: July 31, 1998.

Effective date: July 31, 1998, with full implementation prior to restart from the sixth refueling outage.

Amendment No.: 123.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: February 25, 1998 (63 FR 9600). The March 12 and June 9, 1998, letters provided clarifying information that was within the scope of the original **Federal Register** 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 July 31, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: June 26, 1998 (NRC–98–0040) as supplemented July 16, 1998 (NRC–98–0096), and July 23, 1998 (NRC–98–0117).

Brief description of amendment: The amendment provides a one-time extension of the interval for a number of technical specification (TS) surveillance requirements that will be performed during the sixth refueling outage. TS 4.0.2 and Index page xxii are revised and TS tables 4.0.2–1 and 4.0.2–2 are replaced to reflect the extensions.

NRC has also granted the request of Detroit Edison Company to withdraw a portion of its June 26, 1998, application. By letter dated July 16, 1998, the licensee made some editorial changes and withdrew the portion of the submittal related to TS 4.0.5 for the inservice testing of two valves. A change to the schedule for these valves will be handled within the Inservice Testing Program and a TS change is not necessary. For further details with respect to this action, see the application for amendment dated June 26, 1998, and the licensee's letter dated July 16, 1998, which withdrew this portion of the application for the license amendment, and the staff's safety evaluation enclosed with the amendment. By letter dated July 23, 1998, the licensee added an additional surveillance requirement for two instruments to the amendment. The above documents 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 listed below.

Date of issuance: August 4, 1998.

Effective date: August 4, 1998, with full implementation within 30 days.

Amendment No.: 124.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 2, 1998 (63 FR 36273). The July 16 and July 23, 1998, letters provided clarifying information and updated TS pages that were within the scope of the original **Federal Register** 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 August 4, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

Duke Energy Corporation, 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: July 8, 1998.

Brief description of amendments: The amendments revise TS 4.5.4.1.b.1 for testing the Penetration Room Ventilation System air flow by adding a reference to the following statement that has been added to the bottom of the TS page: ''A temporary noncompliance with this surveillance requirement is allowed until August 30, 1998, to complete necessary modifications to enable flow testing in accordance with ANSI N510–1975.'' This action supersedes the Notice of Enforcement Discretion that was issued by the staff on July 8, 1998.

Date of Issuance: August 7, 1998.

Effective date: As of the date of issuance.

Amendment Nos.: Unit 1—231; Unit 2—231; Unit 3—228.

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

Public comments requested as to proposed no significant hazards consideration: Yes. (63 FR 38433 dated July 16, 1998). 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 August 17, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated August 7, 1998.

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

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

Date of application for amendment: May 19, 1995, as supplemented by letters dated February 27 and September 30, 1996.

Brief description of amendment: The amendment modifies the technical specifications (TSs) to extend the allowed outage times (AOTs) for a single inoperable Safety Injection Tank (SIT) from one hour to 24 hours, and for a single inoperable SIT specifically due to malfunctioning SIT water level or nitrogen cover pressure instrumentation inoperability from one hour to 72 hours.

Date of issuance: August 7, 1998.

Effective date: August 7, 1998.

Amendment No.: 192.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 2, 1995 (60 FR 39439).

The February 27 and September 30, 1996, submittals provided clarifying information that did not change the initial proposed NSHC determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 18, 1998, and supplemented June 30, 1998.

Brief description of amendment: The amendment proposed to revise the Improved Technical Specifications to allow operation with a number of indications previously identified as tube end anolmalies and multiple tube end anolmalies in the Crystal River Unit 3 Once Through Steam Generator tubes.

Date of issuance: July 30, 1998.

Effective date: July 30, 1998.

Amendment No.: 169.

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

Date of initial notice in **Federal Register**: June 30, 1998 (63 FR 35615). The June 30, 1998 supplement included clarifying information which did not change the original no significant hazards determination.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 27, 1998.

Brief description of amendments: The amendments revise the Administrative Controls, Unit Staff Section 6.2.2.f of TS to authorize the use of various controlled shift structures and durations during a nominal (36 to 48 hours) work week. This includes the use of up to 12-hour shifts without heavy use of overtime.

Date of Issuance: July 30, 1998.

Effective Date: July 30, 1998.

Amendment Nos.: 155 and 93.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 1, 1998 (63 FR 35989).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 31, 1996.

Brief description of amendment: The amendment deletes Table 3.5.2 which lists automatic primary containment isolation valves. In addition, the amendment clarifies the applicability of an action statement that applies to several limiting conditions for operation in Section 3.5, and deletes closure time requirements for several automatic isolation valves in Section 4.5.F.

Date of Issuance: August 13, 1998.

Effective date: August 13, 1998, to be implemented within 60 days.

Amendment No.: 196.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 18, 1996 (61 FR 66707).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 13, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 22, 1995.

Brief description of amendment: The amendment changes Technical Specification 5.2.2.e, ''Unit Staff,'' by revising the requirements for controls on the working hours of unit staff who perform safety related functions.

Date of issuance: August 13, 1998.

Effective date: August 13, 1998.

Amendment No.: 115.

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

Date of initial notice in **Federal Register:** December 20, 1995 (60 FR 65681).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 13, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public

Library, 120 West Johnson Street, Clinton, IL 61727.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: January 22, 1998, as supplemented July 17, 1998.

Brief description of amendment: The amendment revises the Millstone Unit 3 licensing basis to accept the existing use of epoxy coatings on safety related components. The revised licensing basis will be incorporated into Chapter 9 of the Final Safety Analysis Report.

Date of issuance: August 7, 1998.

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

Amendment No.: 162.

Facility Operating License No. NPF–49: Amendment revised the Final Safety Analysis Report and the Facility Operating License.

Date of initial notice in **Federal Register:** February 25, 1998 (63 FR 9606).

The July 17, 1998, letter provided clarifying information that did not change the scope of the January 22, 1998, application, and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 24, 1996, as supplemented October 17, 1996, January 3, January 20, and November 10, 1997, and January 9, June 8, and July 20, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant Units 1 and 2 to allow use of an alternate steam generator tube repair criteria (elevated F-star or EF*) in the tubesheet region when used with the repair method of additional roll expansion. The amendments incorporate revised acceptance criteria for tubes with degradation in the tubesheet region and enable the licensee to avoid unnecessary plugging and sleeving of steam generator tubes.

Date of issuance: August 13, 1998.

Effective date: August 13, 1998, with full implementation within 30 days.

Amendment Nos.: 137 and 128.

Facility Operating License Nos. DPR–42 and DPR–60. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 4, 1996 (61 FR 64388).

The licensee's submittals dated January 3, January 20, and November 10, 1997, and January 9, June 8, and July 20, 1998, provided additional clarifying information within the scope of the original **Federal Register** notice and did not affect the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 13, 1998.

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.

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

Date of application for amendment: February 6, 1998.

Brief description of amendment: The amendment revises the Reactor Protection System Normal Supply Electrical Protection Assembly Undervoltage Trip Setpoint.

Date of issuance: July 29, 1998.

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

Amendment No.: 245.

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

Date of initial notice in **Federal Register:** April 22, 1998 (63 FR 19976).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 14, 1997.

Brief description of amendments: The amendments add technical specification (TS) surveillance requirements for the service water accumulator vessels. Specifically, surveillance requirements are provided for vessel level, pressure and temperature, and discharge valve response time. The surveillance requirements are included in TS 3/4.6.1.1 and 3/4.6.2.3, and the applicable Bases sections are expanded to provide supporting information.

Date of issuance: August 6, 1998.

Effective date: August 6, 1998.

Amendment Nos.: 213 and 193.

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

Date of initial notice in **Federal Register:** January 28, 1998 (63 FR 4432).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 6, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 6, 1998.

Brief description of amendment: The requested changes would replace the two percent penalty addressed in Surveillance Requirement 3.2.1.2(a) with a burnup-dependent factor to be specified in the WBN Core Operating Limits Report and makes associated changes to the administrative controls in Technical Specification 5.9.5 and the BASES.

Date of issuance: August 10, 1998.

Effective date: August 10, 1998.

Amendment No.: 11.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** June 17, 1998 (63 FR 33109).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 10, 1998.

No significant hazards consideration comments received: None

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402.

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: November 5, 1997.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Sections 3.9.7, 4.9.7.1, 4.9.7.2, and 3/4.9.7 for Unit 1, and 3.9.7, 4.9.7.1, 4.9.7.2, and 3/4 .9.7 for Unit 2, allowing the movement of the spent fuel pit gate over the irradiated fuel.

Date of issuance: August 3, 1998.

Effective date: August 3, 1998.

Amendment Nos.: 213 and 194.

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

Date of initial notice in **Federal Register:** December 17, 1997 (62 FR 66146).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 1998.

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.

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

Dated at Rockville, Maryland, this 19th day of August 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–22766 Filed 8–25–98; 8:45 am]

BILLING CODE 7590–01–P

POSTAL SERVICE

Information Based Indicia Program (IBIP) Performance Criteria for Information Based Indicia and Security Architecture for IBI Postage Metering Systems (PCIBISAIBIPMS)

AGENCY: Postal Service.

ACTION: Notice of USPS response to public comments and availability of Performance Criteria with request for comments.

SUMMARY: The Postal Service has published a set of draft specifications for the Information-Based Indicia Program (IBIP). In an effort to comply with comments received regarding those specifications we have compiled a set of functional Performance Criteria as defined in this release. The following published specifications are hereby superseded by this Performance Criteria release:

IBIP Open System Indicia Specification dtd July 23, 1997

IBIP Open System PSD Specification dtd July 23, 1997

IBIP Open System Host Specification dtd October 9, 1996

IBIP Key Management Plan dtd April 25, 1997

The Postal Service also seeks comments on intellectual property issues raised by IBIP Performance Criteria, policy, and procedures if adopted in present form. If an intellectual property issue includes patents or patent applications covering any implementations of the Performance Criteria, the comment should include a listing of such patents and applications and the license terms available for such patents and applications.

ADDRESSES: Copies of the Performance Criteria noted above may be obtained from Edmund Zelickman, United States Postal Service, 475 L'Enfant Plaza SW, Room 1P–801, Washington DC 20260–2444. Copies of all written comments may be inspected, by appointment, between 9 a.m. and 4 p.m., Monday through Friday, at the above address.

DATES: All written comments must be received on or before October 26, 1998.

FOR FURTHER INFORMATION CONTACT: Edmund Zelickman at (202) 268–3940.

Stanley F. Mires,

Chief Counsel, Legislative.

[FR Doc. 98–22923 Filed 8–25–98; 8:45 am]

BILLING CODE 7710–12–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549

Extension:

Rule 17 Ad–6, SEC File No. 270–151, OMB Control No. 3235–0291

Rule 17 Ad–7, SEC File No. 270–152, OMB Control No. 3235–0136

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission ("Commission") has submitted to the Office of Management and Budget requests for extension of previously approved collections of information discussed below:

- **Rule 17Ad–6 Recordkeeping Requirements for Transfer Agents**

Rule 17 Ad–6 under the Securities Exchange Act of 1934 (15 U.S.C. § 78b *et seq.*) requires every registered transfer agent to make and keep current records about a variety of information, such as: (1) specific operational data regarding the time taken to perform transfer agent activities (to ensure compliance with the minimum performance standards in Rule 17Ad–2 (17 CFR 240.17Ad–2)); (2) written inquiries and requests by shareholders and broker-dealers and response time thereto; (3) resolutions, contracts or other supporting documents concerning the appointment or termination of the transfer agent; (4) stop orders or notices of adverse claims to the securities; and (5) all canceled registered securities certificates.

These recordkeeping requirements ensure that all registered transfer agents are maintaining the records necessary to monitor and keep adequate control over their own performance and to examine registered transfer agents on an historical basis for compliance with applicable rules.

It is estimated that approximately 1,248 registered transfer agents will spend a total of 599,040 hours per year complying with Rule 17Ad–6. Based on average cost per hour of $50, the total cost of compliance with Rule 17Ad–6 is $29,952,000.

The retention period for the recordkeeping requirement under Rule 17Ad–6 is six months to one year. In addition, such records must be retained for a total of two to six years or for one year after termination of the transfer agency, depending on the particular record or document. The recordkeeping requirement under Rule 17Ad–6 is mandatory to assist the Commission and other regulatory agencies with monitoring transfer agents and ensuring compliance with the rule. This rule does not involve the collection of confidential information.

- **Rule 17Ad–7 Recordkeeping Requirements for Transfer Agents**

Rule 17Ad–7 under the Securities Exchange Act of 1934 (15 U.S.C. § 78b *et seq.*) requires each registered transfer agent to retain, in an easily accessible place for a period of six months to one year, all the records required to be made and kept current under the Commission's rules regarding registered transfer agents. Rule 17Ad–7 also requires such records to be retained for a total of two to six years or for one year after termination of the transfer agency, depending on the particular record or document.