present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, its consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Dr. Medhat El-Zeftawy (telephone 301/415–6889) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes in the proposed agenda, etc., that may have occurred.

Date: September 27, 2000.

James E. Lyons,

Associate Director for Technical Support ACRS/ACNW.

[FR Doc. 00–25461 Filed 10–3–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of October 2, 9, 16, 23, 30, and November 6, 2000.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of October 2

Friday, October 6

9:25 a.m.
Affirmation Session (Public Meeting) (If needed)

9:30 a.m.
Meeting with ACRS (Public Meeting) (Contact: John Larkins, 301–415–7360)

This meeting will be webcast live at the Web address—

www.nrc.gov/live.html

Week of October 9—Tentative

There are no meetings scheduled for the Week of October 9.

Week of October 16—Tentative

Tuesday, October 17

9:25 a.m.
Affirmation Session (Public Meeting) (If needed)

Week of October 23—Tentative

Monday, October 23

1:55 p.m.
Affirmation Session (Public Meeting) (If needed)

Week of October 30—Tentative

There are no meetings scheduled for the week of October 30.

Week of November 6—Tentative

There are no meetings scheduled for the Week of November 6.

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (Recording)—)(301) 415–1292. Contact Person for More Information: Bill Hill (301) 415–1661.

The NRC Commission Meeting Schedule can be found on the Internet at:

http://www.nrc.gov/SECY/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: September 29, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–25564 Filed 10–2–00; 11:36 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

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 September 11, 2000, through September 22, 2000. The last biweekly notice was published on September 20, 2000 (65 FR 56946, as corrected at 65 FR 57484 and 65 FR 58113).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By November 3, 2000, 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 NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the NRC's Public Document Room, One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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 NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of amendments request: June 8, 2000.

Description of amendments request: The licensee proposes to amend Technical Specification (TS) 5.6.5, ''Core Operating Limits Report (COLR),'' to add a methodology using the

CASMO–4 and SIMULATE–3 codes to the list of analytical methods used to determine core operating limits contained in TS 5.6.5.b. The change would allow the use of the CASMO–4 and SIMULATE–3 methodology to perform nuclear design calculations.

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:

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

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Arizona Public Service Company (APS) intends to replace the DIT/ROCS/MC methodology with CASMO–4/SIMULATE–3 code package. The proposed amendment would add methodology using CASMO–4 and SIMULATE–3 codes to the list of analytical methods used to determine core operating limits contained in Technical Specification 5.6.5.b. This will allow the use of the CASMO–4 and SIMULATE–3 methodology to perform all steady-state PWR [pressurized-water reactor] core physics analyses.

The probability of occurrence of an accident previously evaluated will not be increased by the proposed change in the particular codes used for physics calculations for nuclear design analysis. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the Updated Final Safety Analysis Report (UFSAR). These inputs do not alter the physical characteristics or modes of operation of any system, structure, or component involved in the initiation of an accident. Thus, there is no significant increase in the probability of an accident previously evaluated as a result of this change.

The consequences of an accident evaluated in the UFSAR are affected by the value of inputs to the transient safety analysis. An extensive benchmark of CASMO–4/SIMULATE–3 predictions with measured data using a variety of fuel designs and operating conditions in power reactors and critical experiments, was performed. The accuracy of CASMO–4/SIMULATE–3 is similar to, and sometimes better than, the accuracy of DIT/ROCS/MC. Furthermore, there is always the potential for the value of the nuclear design parameters to change solely as a result of the new reload fuel core loading pattern. Regardless of the source of a change, an assessment is always made of changes to the nuclear design parameters with respect to their effects on the consequences of accidents previously evaluated in the UFSAR. Refueling is an anticipated activity which is described in the UFSAR. If increased consequences are anticipated, compensatory actions are implemented to neutralize any expected increase in consequences. These compensatory actions include, but are not limited to, crediting any existing margins in the analysis or redefining the operating envelope to avoid increased consequences. Thus, the nuclear design parameters are intermediate results and by themselves will not result in an increase in the consequence of an accident evaluated in the UFSAR.

Therefore, the replacement of the DIT/ROCS/MC codes with the CASMO–4/SIMULATE–3 code package, which will perform the same functions as the DIT/ROCS/MC codes with similar accuracy, does not significantly increase the consequences of an accident previously evaluated.

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

No. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Arizona Public Service Company (APS) intends to replace the DIT/ROCS/MC methodology with CASMO–4/SIMULATE–3 code package. The proposed amendment would add methodology using CASMO–4 and SIMULATE–3 codes to the list of analytical methods used to determine core operating limits contained in Technical Specification 5.6.5.b.

The possibility for a new or different kind of accident evaluated previously in the UFSAR will not be created by the proposed change to the particular codes used for physics calculations for nuclear design analyses. The change involves replacing the NRC approved ABB Combustion Engineering Nuclear Power (ABB/CE) DIT and ROCS/MC codes, with the Studsvik CASMO–4 and SIMULATE–3 codes. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the UFSAR. These inputs do not alter the physical characteristics or modes of operation of any system, structure or component involved in the initiation of an accident.

Therefore, the replacement of the DIT/ROCS/MC codes with the CASMO–4/SIMULATE–3 code package, which will perform the same functions as the DIT/ROCS/MC codes with similar accuracy, does not increase the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3—Does the proposed change involve a significant reduction in a margin of safety?

No. The proposed change does not involve a significant reduction in a margin of safety.

The margin of safety as defined in the basis for any technical specification will not be reduced nor increased by the proposed change to the particular codes used for physics calculations for nuclear design analyses. The change involves replacing the NRC approved ABB/CE DIT and ROCS/MC codes, with the Studsvik CASMO–4 and SIMULATE–3 codes. Extensive benchmarking of the CASMO–4/SIMULATE–3 computer codes has demonstrated that the values of those parameters used in the safety analysis are not significantly changed relative to the values obtained using the DIT/ROCS/MC computer codes. For any changes in the calculated values that do occur, the application of appropriate biases and uncertainties ensures that the current margin of safety is maintained. Specifically, use of these code specific biases and uncertainties in safety evaluations continues to provide the same statistical assurance that the values of the nuclear parameters used in the safety analysis are conservative with respect to the actual values on at least a 95/95 probability/confidence basis.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: June 16, 2000.

Description of amendments request: The licensee proposes to amend Technical Specification (TS) Table 3.3.10–1, ''Post Accident Monitoring Instrumentation,'' to add the High Pressure Safety Injection cold leg flow and hot leg flow instrumentation to this table. This change is required because this instrumentation meets the criteria for a Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident,'' Revision 2, Type A, Category 1 variable.

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:

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

No. The proposed change to TS Table 3.3.10–1, by adding the High Pressure Safety Injection (HPSI) hot and cold leg flow instrumentation, does not involve a significant increase in the probability or consequences of an accident previously evaluated because it does not represent a change to design configuration or operation of the plant. The amendment does not affect the operability or availability of the HPSI system or any other safety related equipment. Additionally, there are no effects on the failure modes associated with the probability of a failure of a system important to safety.

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

No. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated because the change does not impact the response or operation of the plant. The availability and operability of the plant equipment is unchanged, as the design requirements have not changed.

The proposed change revises only the Regulatory Guide (RG) 1.97, Revision 2, classification of the HPSI hot leg and cold leg flow indication loops. Regardless of RG classification the instruments remain seismically, electrically, and otherwise qualified for the application. Hence, the revised classification will not subject these components to new modes of operation that could result in a new failure mode, thus initiating an accident of a different type.

Standard 3—Does the proposed change involve a significant reduction in a margin of safety?

No. This proposed amendment does not involve a significant reduction in the margin of safety because neither of the following PVNGS Technical Specification (TS) Bases (B 3.5.3 ECCS [emergency core cooling system]—Operating, or 3.3.10 Post Accident Monitoring (PAM) Instrumentation) is changed by the proposed amendment.

TS Bases B 3.5.3 ECCS—Operating—states that the function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of Coolant Accident (LOCA);

b. Control Element Assembly (CEA) ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam Generator Tube Rupture (SGTR).

Changing the RG 1.97 Type and Category of these instruments does not affect the ability of the ECCS to provide core cooling and negative reactivity during these accidents.

TS Bases B 3.3.10—Post Accident Monitoring (PAM) Instrumentation—states that the primary purpose of PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Events.

The OPERABILITY of PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

These Type A variables are required to be included in this LCO [Limiting Condition for Operation] because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). The addition of these instruments supports this TS Bases. 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 that review, it appears that the three 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.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 5, 1999, as supplemented on December 22, 1999, and September 18, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.5, ''Instrumentation Systems,'' for the reactor protection system and engineered safety features actuation system instrumentation. Specifically, the proposed amendment would (1) change the allowed outage times for the instrumentation and the analog channel test bypass time and (2) allow on-line testing and maintenance of instrumentation. The proposed amendment also includes several editorial changes to TS Tables 3.5–2 and 3.5–3. The proposed amendment was originally noticed in the **Federal Register** on September 8, 1999 (64 FR 48861). It is now being noticed to correct errors made in the original notice description of amendment request.

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 reactor protection and engineered safety features functions are not initiators of any design basis accident or event and therefore do not increase the probability of any accident previously evaluated. The proposed changes to the AOTs [allowed outage times], bypass times, and allowing on-line testing and maintenance have an insignificant impact on plant safety based on the calculated CDF [core damage frequency] increase being less than 1.0E–06. Therefore, the proposed changes do not result in a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do not result in a change in the manner in which the RPS [reactor protection system] and ESFAS [engineered safety features actuation system] provide plant protection. No change is being made which alters the functioning of the RPS and ESFAS. Rather, the likelihood or probability of the RPS or ESF functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident nor involve a reduction in the margin of safety as defined in the Safety Analysis Report.

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

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operations are determined. The impact of increased AOTs, testing times, and allowing on-line testing and maintenance are expected to result in an overall improvement in safety because:

The longer AOTs for the master relays, logic cabinets, and analog channels will promote improved maintenance practices that will provide improved component performance, improved availability of the protection system, and a reduced number of spurious reactor trips and spurious actuation of safety equipment.

The longer AOTs and bypass times for the analog channels will provide additional time before being required to place the channel in trip. With the channel in trip, the logic required to cause a reactor trip or a safety system actuation is reduced to 1 of 2 (for 2 of 3 logic) and to 1 of 3 (for 2 of 4 logic). With the reduced logic requirement, the potential for a spurious actuation is increased. Leaving the channel in the bypass state for additional time does reduce the availability of signals to initiate component actuation for event mitigation when required, but as shown in this analysis, the impact on plant safety is small due to the availability of other signals or operator action to trip the reactor or cause component actuation.

The longer allowed outage times will provide plant operators additional flexibility in operating the plant. There will be additional time available before an action needs to be taken to shut down the plant or place a channel in the tripped state. This additional flexibility will facilitate prioritizing component repairs.

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.

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

NRC Section Chief: Marsha Gamberoni.

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 amendment request: September 12, 2000.

Description of amendment request: The proposed amendments would revise the Technical Specifications 5.5.10, Item e.6, Steam Generator Tube Surveillance Program, by (1) removing the restriction on the lower tube sheet area rolling, (2) removing the limitation of only one reroll per steam generator tube, (3) eliminating the requirement that the reroll be one inch in depth, and (4) changing the revision number reference for Topical Report BAW–2303P, August 2000, ''OTSG Repair Roll Qualification Report,'' from Revision 3 to Revision 4.

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:

Duke Energy Corporation (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by NRC regulations in 10CFR50.92. This ensures operation of the facility in accordance with the proposed amendment will not:

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

No. The proposed change to the Technical Specifications incorporates Revision 4 of Topical Report BAW–2303P, OTSG Repair Roll Qualification Report. This document is also being submitted for NRC review and approval. This revision addresses, and is consistent with, the conclusions of all applicable Oconee licensing basis analyses and ensures that previously evaluated accidents are bounding. All the established acceptance criteria for the accidents analyzed in the Oconee licensing basis continue to be met. Therefore, no existing accident probabilities or consequences will be impacted.

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

No. Revision 4 of BAW–2303P addresses limiting events for steam generator tube reroll repairs. These events include Main Steam Line Break, the Small Break Loss of Coolant Accident, and other transients on B&W Once-Through Steam Generators. For Oconee, the Main Steam Line Break is the limiting event. This revised topical report confirms the acceptability of the reroll repair techniques previously used at Oconee. As a result, no new failure modes are being created. BAW–2303P, as submitted for NRC review and approval, does not create the possibility of a new or different kind of accident.

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

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. As part of the reactor coolant system pressure boundary, the steam generator tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems, such that residual heat can be removed from the primary system. In addition, the steam generator tubes also isolate the radioactive fission products in the primary coolant from the secondary system. Finally, the steam generator tubes may be relied upon to maintain their integrity under conditions resulting from core damage severe accidents consistent with the containment objectives of preventing uncontrolled fission product release. The functions of the steam generator tubes will not be significantly affected by the changes proposed in this license amendment request. Implementation of BAW–2303P, Revision 4, as submitted for NRC review and approval, at Oconee will result in assurance that parameters affecting the integrity of the steam generator tubes continue to meet applicable safety analyses and industry codes and standards. Therefore, no safety margin will be significantly reduced.

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.

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard L. Emch, Jr.

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 amendment request: July 26, 2000.

Description of amendment request: The amendments would revise the Technical Specifications (TS) Index to delete reference to the Bases since, in accordance with 10 CFR 50.36(a), the Bases are not a part of the TS. Future changes to the TS Bases will be evaluated per 10 CFR 50.59 and made under administrative control and reviews and in accordance with the proposed TS Bases control program as described in TS 5.5.14 of NUREG–1432, Revision 1, ''Standard Technical Specifications Combustion Engineering Plants.''

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

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

The proposed amendments are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. The Technical Specification BASES, per 10 CFR 50.36(a), are not part of the Technical Specifications. Changes to the TS BASES will be controlled by a plant procedure under administrative controls and reviews. Proposed changes to the TS BASES will be evaluated in accordance with 10 CFR 50.59 and made under the programmatic controls and requirements of the proposed Technical Specifications (TS) Bases Control Program. Therefore, the proposed changes do not increase the probability or consequences of accidents previously analyzed.

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

The proposed amendments are administrative in nature. The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative change, since the proposed change does not involve the addition or modification of equipment nor does it alter the design or operation of affected plant systems, structures, or components.

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

The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The BASES information, per 10 CFR 50.36(a), is not a part of the Technical Specifications. Changes to the TS BASES will be controlled by a plant procedure under administrative controls and reviews and made under the programmatic controls and requirements of the proposed Technical Specifications (TS) Bases Control Program. Proposed changes to the TS BASES will be evaluated in accordance with 10 CFR 50.59 and the TS BASES will be maintained in an FPL-controlled document. Therefore, the proposed changes do not reduce any margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: August 31, 2000.

Description of amendment request: This amendment would revise Improved Technical Specification (ITS) Table 3.3.18–1, ''Remote Shutdown System Instrumentation.'' The table would be updated to reflect plant modifications and procedure changes regarding placing and maintaining the plant in a safe shutdown condition if the control room becomes inaccessible.

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

The proposed changes do not involve a significant hazards consideration. In support of this conclusion, the following analysis is provided:

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

The instruments listed in Table 3.3.18–1 are used to provide information on selected parameters to the operators that will allow them to place and maintain the plant in a safe shutdown condition in the event the control room becomes inaccessible. The proposed license amendment revises Table 3.3.18–1, Remote Shutdown System Instrumentation, to more accurately reflect the instruments that would be used by the operators to perform abnormal operating procedure AP–990, Shutdown from Outside the Control Room. The proposed license amendment also revises ITS Bases Section B 3.3.18 to add a table that identifies, by equipment tag number, the specific instruments used to satisfy the requirements of ITS 3.3.18 and ITS Table 3.3.18–1. The instruments identified in ITS Table 3.3.18–1 and ITS Bases Table B 3.3.18–1 are not initiators of any design basis accidents. The design functions of the Remote Shutdown System Instrumentation and the initial conditions for accidents that require the Remote Shutdown System will not be effected by the change. Therefore, the change will not increase the probability or consequences of an accident previously evaluated.

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

The proposed amendment involves no changes to the CR–3 design or to the functions or operation of the Remote Shutdown System. The proposed amendment will ensure that sufficient and appropriate instrumentation is available to allow the operators to place and maintain the plant in a safe shutdown condition in the event the control room becomes inaccessible. The proposed amendment will also add information to Bases Section B 3.3.18 that will ensure timely and accurate operability evaluations and entry into the appropriate Conditions and Required Actions of ITS 3.3.18. The proposed amendment will not create any new plant configurations different from those already analyzed. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does not involve a significant reduction in the margin of safety.

The proposed amendment revises Table 3.3.18–1, Remote Shutdown System Instrumentation, to more accurately reflect the instruments that would be used by the operators to perform a shutdown from outside the control room. The proposed amendment will revise ITS Bases Section B 3.3.18 to provide the operators with guidance that will assist them in making timely and accurate operability determinations and entries into the appropriate Conditions and Required Actions for ITS 3.3.18. The proposed changes will not reduce the ability of the Remote Shutdown System to monitor and control reactivity, RCS [reactor coolant system] pressure, core heat removal, or RCS inventory. Thus, the proposed amendment will not result in 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.

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC—A5A, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Richard P. Correia.

Pacific Gas and Electric Company, Docket No. 50–323, Diablo Canyon Nuclear Power Plant, Unit No. 2, San Luis Obispo County, California

Date of amendment requests: June 19, 2000.

Description of amendment requests: The proposed license amendment would revise Technical Specifications 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' and 5.6.10, ''SG Tube Inspection Report,'' to add new surveillance and reporting requirements associated with a SG tube inspection and repair. The new requirements establish alternate repair criteria for axial primary water stress corrosion cracking at dented tube support plate intersections.

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.

Examination of crack morphology for primary water stress corrosion cracking (PWSCC) at dented intersections has been found to show one or two microcracks well aligned with only a few uncorroded ligaments and little or no other inside diameter axial cracking at the intersection. This relatively simple morphology is conducive to obtaining good accuracy in nondestructive examination (NDE) sizing of these indications. Accordingly, alternate repair criteria (ARC) is established based on crack length and average and maximum depth within the thickness of the tube support plate (TSP).

The application of the ARC requires a Monte Carlo condition monitoring assessment to determine the as-found condition of the tubing. The condition monitoring analysis described in WCAP–15128 Revision 3 is consistent with NRC Generic Letter 95–05 requirements.

The application of the ARC requires a Monte Carlo operational assessment to determine the need for tube repair. The repair bases are obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile using Monte Carlo analysis techniques described in WCAP–15128 Revision 3. The burst pressure and leakage is compared to the requirements in WCAP–15128 Revision 3. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected end of cycle (EOC) requirements are satisfied, the tube will be left in service.

A steam generator (SG) tube rupture event is one of a number of design basis accidents that are analyzed as part of a plant's licensing basis. A single or multiple tube rupture event would not be expected in a SG in which the ARC has been applied. The ARC requires repair of any indication having a maximum crack depth greater than or equal to 40 percent outside the TSP, thus limiting the potential length of a deep crack outside the TSP at EOC conditions and providing margin against burst and leakage for free span indications.

For other design basis accidents such as a main steam line break, main feed line break, control rod ejection, and locked reactor coolant pump motor, the tubes are assumed to retain their structural integrity.

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.

Implementation of the proposed SG tube ARC does not introduce any significant changes to the plant design basis. A single or multiple tube rupture event would not be expected in a SG in which the ARC has been applied. Both condition monitoring and operational assessments are completed as part of the implementation of ARC to determine that structural and leakage margin exists prior to returning SGs to service following inspections. If the condition monitoring requirements are not satisfied for burst or leakage, the causal factors for EOC indications exceeding the expected values will be evaluated. The methodology and application of this ARC will continue to ensure that tube integrity is maintained during all plant conditions consistent with the requirements of Regulatory Guide (RG) 1.121 and Revision 1 of RG 1.83. Therefore, a permanent ARC is justified.

In the analysis of a SG tube rupture event, a bounding primary-to-secondary leakage rate equal to the operational leakage limits in the Technical Specifications (TS), plus the leak rate associated with the double ended rupture of a single tube, is assumed. For other design basis accidents, the tubes are assumed to retain their structural integrity and exhibit primary-to-secondary leakage within the limits assumed in the current licensing basis accident analyses. Steam line break leakage rates from the proposed PWSCC ARC are combined with leakage rates from other approved ARC (*i.e.*, voltage-based ARC and W* ARC). The combined leakage rates will not exceed the limits assumed in the current licensing basis accident analyses.

The 40 percent maximum depth repair limit for free span indications provides a very low likelihood of free span leakage under design basis or severe accident conditions. Leakage from indications inside the TSP is limited by the constraint of the TSP even under severe accident conditions, and leakage behavior in a severe accident would be similar to that found acceptable by the NRC under approved ARC for axial outside diameter stress corrosion cracking (ODSCC) at TSP intersections. Therefore, even under severe accident conditions, it is concluded that application of the proposed ARC for PWSCC at dented TSP locations results in a negligible difference in risk of a tube rupture or large leakage event, when compared to current 40 percent repair limits or previously approved ARC.

DCPP continues to implement a maximum operating condition leak rate limit of 150 gallons per day per SG to preclude the potential for excessive leakage during all plant conditions.

The possibility of a new or different kind of accident from any previously evaluated is not created because SG tube integrity is maintained by inservice inspection, condition monitoring, operational assessment, tube repair, and primary-to-secondary leakage monitoring.

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

Tube repair limits provide reasonable assurance that tubes accepted for continued service without plugging or repair will exhibit adequate tube structural and leakage integrity during subsequent plant operation. The implementation of the proposed ARC is demonstrated to maintain SG tube integrity consistent with the criteria of draft NRC Regulatory Guide 1.121. The guidelines of RG 1.121 describe a method acceptable to the NRC staff for meeting General Design Criteria (GDC) 2, 4, 14, 15, 31, and 32 by ensuring the probability or the consequences of SG tube rupture remain within acceptable limits. This is accomplished by determining the limiting conditions of degradation of SG tubing, for which tubes with unacceptable cracking should be removed from service.

Upon implementation of the proposed ARC, even under the worst-case conditions, the occurrence of PWSCC at the tube support plate elevations is not expected to lead to a SG tube rupture event during normal or faulted plant conditions. The ARC involves a computational assessment to be completed for each indication left in service ensuring that performance criteria for tube integrity and leak tightness are met until the next scheduled outage. Therefore, a permanent ARC is justified.

As discussed below, certain tubes are excluded from application of ARC. Existing tube integrity requirements apply to these tubes, and the margin of safety is not reduced.

In addressing the combined loading effects of a loss-of-coolant (LOCA) and safe shutdown earthquake (SSE) on the SGs (as required by GDC 2), the potential exists for yielding of the TSP in the vicinity of the wedge groups, accompanied by deformation of tubes and a subsequent postulated in-leakage. Tube deformation could lead to opening of pre-existing tight through wall cracks, resulting in secondary to primary in-leakage following the event, which could have an adverse affect on the Final Safety Analysis Report (FSAR) results. Based on a DCPP analysis of LOCA and SSE, SG tubes located in wedge region exclusion zones are susceptible to deformation, and are excluded from application of ARC.

A DCPP tube stress analysis for feed line break (FLB)/steam line break (SLB) plus SSE loading determined that high bending stresses occur in certain SG tubes at the seventh TSP, because the stresses exceed the maximum imposed bending stress for existing test data (equal to approximately the lower tolerance limit yield stress). These tubes are located in rows 11 to 15 and 36 to 46, and are excluded from application of ARC.

Tube intersections that contain TSP ligament cracking are also excluded from application of ARC.

Based on the above, it is concluded that the proposed license amendment requires does not result in a significant reduction in margin [of safety] with respect to the plant safety analyses as defined in the FSAR or TS.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: August 28, 2000.

Description of amendment request: The proposed amendment would revise the Units 1, 2 and 3 Technical Specifications (TS) to incorporate Technical Specifications Task Force (TSTF) Nos.TSTF–71, TSTF–208, TSTF–222, TSTF–284, TSTF–258 and TSTF–364. TSTFs are changes that were submitted to the staff by the nuclear power industry TSTF that have generic applicability. A description of each of the six TSTFs follows: (1) TSTF–71, Revision 2, adds an example of the application of the Safety Function Determination Program (SFDP) to the Bases for Limiting Conditions for Operation (LCO) 3.0.6. (2) TSTF–208, Revision 0, extends the allowed time to reach MODE 2 in LCO 3.0.3 from 7 hours to 10 hours. The change is based on plant experience regarding the time needed to perform a controlled shutdown in an orderly manner. (3) TSTF–222, Revision 1, clarifies Improved Technical Specification (ITS) Section 3.1.4, Control Rod Scram Times, Surveillance Requirements (SRs) to better delineate the requirements for testing control rods following refueling outages and for control rods requiring testing due to work activities. (4) TSTF–258, Revision 4, revises TS Section 5.0, Administrative Controls, to delete specific TS staffing requirement provisions for Reactor Operators (ROs), eliminates TS details for working hour limits, clarifies requirements for the Shift Technical Advisor (STA) position, adds regulatory definitions for Senior ROs and ROs, revises the Radioactive Effluent Controls Program to be consistent with the intent of 10 CFR Part

20, deletes periodic reporting requirements for mainsteam relief valve openings, and revises radiological area control requirements for radiation areas to be consistant with those specified in 10 CFR 20.1601(c). (5) TSTF–284, Revision 3, modifies Improved ITS Section 1.4, Frequency, to clarify the usage of the terms ''met'' and ''performed'' to facilitate the application of SR Notes. Two new SR Examples, 1.4–5 and 1.4–6, are added to illustrate the application of the terms. (6) TSTF–364, Revision 0, revises Section 5.5.10, TS Bases Control Program, to reference 10 CFR 50.59 rather than ''unreviewed safety question.'' Also, editorial change WOG–ED–24, which substitutes ''require'' for ''involve'' in 5.5.10.b is made for consistency in usage.

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, which are presented below:

(1) TSTF–71, Revision 2.

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

The change adds an example of SFDP use to facilitate the application of the TS, which serves to improve TS usefulness. The proposed change is an administrative clarification of existing requirements, and does not change TS requirements. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. The proposed change will not impose any new or eliminate any existing requirements. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

(2) TSTF–208, Revision 0.

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change relaxes the Action time for LCO 3.0.3. The subject Action time is not an initiating condition for any accident previously evaluated and the accident analyses do not assume that equipment is out of service (requiring entry into LCO 3.0.3) prior to postulated events. Consequently, the extended action time does not significantly increase the probability of an accident previously evaluated. The consequences of an analyzed accident during the extended action time are the same as the consequences during the existing action time. As a result, the consequences of an accident previously evaluated are not significantly increased.

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

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. The TS defines specific time limits during which operation with degraded condition is permitted. In this case, actual plant experience indicates that the Action time in existing TS is too short to accomplish the specified action to be in MODE 2 in an orderly manner. Extension of the time would allow the reactor to be shutdown in a controlled manner while minimizing risks associated with the initiation of inadvertent transients. This maximizes reactor safety. For these reasons, the proposed amendment does not involve a significant reduction in the margin of safety.

(3) TSTF–222, Revision 1.

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

The proposed change is an administrative clarification of existing TS requirements which clarifies scram time testing requirements for control rods. The rewording and reformatting involves no technical changes to the existing TS. As such, there is no effect on initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

(4) TSTF–258, Revision 4.

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

The proposed change is an administrative clarification of existing TS requirements which clarifies and modifies administrative controls in the areas of operator staffing requirements, working hour limits, STA position, Radioactive Effluent Controls Program, periodic reporting requirements for relief valve openings, and radiological control requirements. These TS revisions do not affect analysis inputs for analyzed accidents and transients. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes are administrative type revisions and do not reduce a margin of safety because they have no effect on any safety analyses assumptions. For these reasons, the proposed amendment does not involve a significant reduction in the margin of safety.

(5) TSTF–284, Revision 3.

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

The proposed change is an administrative clarification of existing requirements. The change clarifies the TS terminology to facilitate the use and application of Surveillance Requirement Notes to improve TS use. Also, two additional examples of the application of Surveillance Requirement Notes are incorporated. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. The proposed change will not impose any new or eliminate any existing requirements. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature.

For these reasons, the proposed amendment does not involve a significant reduction in the margin of safety.

(6) TSTF–364, Revision 0.

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

The proposed change is an administrative modification of existing TS requirements for the TS Bases change program to simply reference changes pursuant to 10 CFR 50.59 rather than ''unreviewed safety question''. This change is administrative and has no affect on the current review and approval process for Final Safety Analyses Report and Bases changes. As such, there is no effect on initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change modifies [sic] is an administrative modification of existing TS requirements for the TS FSAR and Bases change program to simply reference changes pursuant to 10 CFR 50.59 rather than ''unreviewed safety question.'' This change is administrative and has no affect on the current review process for FSAR and Bases changes, and will not reduce a margin of safety because it has no effect on any safety analyses assumptions.

For these reasons, the proposed amendment does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 31, 2000 (TS 00–05).

Brief description of amendments: The proposed amendment would revise the Sequoyah Nuclear Plant (SQN) Technical Specifications (TSs). The revision would relocate certain specifications related to reactivity control that are not required to be contained in the TSs by NRC regulations. These specifications include TSs 3.1.2.1 and 3.1.2.2 for boration flow paths, TSs 3.1.2.3 and 3.1.2.4 for boration charging pumps, TSs 3.1.2.5 and 3.1.2.6 for borated water sources, TS 3.1.3.3 for position indication systems during shutdown, and TS 3.10.5 for special test exceptions for the position indication system. These specifications will be relocated in their entirety to the SQN Technical Requirements Manual without changing the requirements currently contained in the TSs.

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

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

The proposed revision relocates the boration specifications and one rod position indication system specification without a change to the requirements and deletes a special test exception for rod position indication that is no longer applicable to SQN. Relocation to the TRM continues to provide an acceptable level of applicability to plant operation and requires revisions to be processed in accordance with the provisions in 10 CFR 50.59. Evaluations of revisions in accordance with 10 CFR 50.59 will continue to ensure that these specifications adequately control the functions for boration and rod position indication systems to maintain safe operation of the plant. The boration systems and the rod position indication system is not postulated to be the initiator of a design basis accident. Since there are no changes to these functions and their operation will remain the same, the probability of an accident is not increased by relocating these requirements to the TRM. Additionally, the accident mitigation capability and offsite dose consequences associated with accidents will not change because these functions will not be altered by the proposed relocation. Therefore the consequences of an accident are not increased by this relocation to the TRM and the control of revisions to these specifications in accordance with 10 CFR 50.59.

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

The proposed revision will not alter the functions for the boration or the rod position indication systems such that accident potential would be changed. The location of these specifications in the TRM and the performance of revisions in accordance with 10 CFR 50.59 will continue to maintain acceptable operability requirements. Therefore, the possibility of an accident of a new or different kind is not created by the proposed relocation and deletion.

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

The proposed specification relocation and special test deletion will not affect plant setpoints or functions that maintain the margin of safety. This is based on the relocation to the TRM continuing to maintain the same level of operability requirements and surveillance testing to adequately ensure functionality of the boration and rod position indication systems. Control of TRM requirements in accordance with 10 CFR 50.59 will ensure that revisions to these functions will not inappropriately impact the health and safety of the public without prior review and approval by NRC. Therefore, the proposed relocation and deletion is acceptable and will not reduce the margin of safety.

The NRC 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.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: May 2, 2000, as supplemented by letter dated August 30, 2000.

Brief description of amendment: The proposed license amendment would obtain approval from the Nuclear Regulatory Commission (NRC) of changes to the Comanche Peak Steam Electric Station, Units 1 and 2 (CPSES)

Security Plans prior to their implementation. Prior approval is being requested from the NRC, in accordance with the requirements of 10 CFR 50.54(p)(1), because some of the changes could have the potential of reducing the effectiveness of the security plans.

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?

Response: No

The proposed changes involving Security activities, do not reduce the ability for the Security organization to prevent radiological sabotage and therefore does not increase the probability or consequences of a radiological release previously evaluated.

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

Response: No

The proposed changes involve functions of the Security organization concerning intrusion detection, material search requirements, alarm response and compensation, and vehicle control. Analysis of the proposed changes has not indicated nor identified a new or different kind of accident previously evaluated.

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

Response: No

Analysis of the proposed changes show that the proposed changes affect only the functions of the Security organization and have no impact upon nor cause a significant reduction in margin of safety for plant operation. The failure points of key safety parameters are not affected.

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.

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

NRC Section Chief: Robert A. Gramm

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

Date of amendment request: September 15, 2000 (WO 00–0036).

Description of amendment request: The proposed amendment would revise footnotes (b) and (c) to Table 1.1–1, ''Modes,'' of the Wolf Creek Technical Specifications, and allow one reactor pressure vessel (RPV) head closure bolt to not be fully tensioned in Mode 4 (hot shutdown) and Mode 5 (cold shutdown). Each RPV head closure bolt is composed of a stud, nut, and washer, and the bolts attach the vessel head to the vessel body. The proposed revisions would allow the plant (1) to be in Modes 4 and 5 with only 53 of 54 RPV head closure bolts fully tensioned (*i.e.,* to allow one head closure bolt to not be fully tensioned), and (2) to operate with one RPV head closure bolt less than fully tensioned. The proposed revision to footnote (b) requires the proposed revision to the definition of refueling (*i.e.,* footnote (c) from the current ''one or more'' RPV head closure bolts detensioned to the proposed ''two or more'' RPV head closure bolts detensioned). In refueling, the RPV head closure bolts are detensioned and removed from the vessel body, and the RPV head is removed from the vessel. The licensee committed to the following program before operating with a not fully tensioned RPV head closure bolt: (1) The circumstances for the closure bolt not being fully tensioned will be reviewed to determined that the analysis in the application is still applicable, (2) the RPV will not be subject to hydrostatic test conditions before the closure bolt is returned to service, and (3) the plant heatup rate will be held to 50°F per hour (half the normal rate) until the closure bolt is returned to service.

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.

Overall protection system performance will remain within the bounds of the accident analyses, since no hardware changes are proposed. Since the stresses [in the RPV head and body] remain within [the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME)] Code allowables, the proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety related equipment assumed to function during an accident situation.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

There are no hardware changes nor are there any changes in the method by which any safety related plant system performs it[s] safety function. The method of plant operation is unaffected. Leakage would be precluded since, as noted in the evaluation [in the application dated September 15, 2000,] adequate compression [of the RPV head closure bolts] remains. However, if leakage were to result from having less than the total number of closure studs fully tensioned it would be detected by an increase in the temperature on the leak-off line from the annular space between the inner and outer vessel head o-rings. That temperature increase would be detected by installed temperature indicators and alarmed in the control room. Any leakage would be detected as an increase in RCS [reactor coolant system] identified LEAKAGE. Since stresses remain within Code allowables, no new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change.

Therefore, the proposed change will not create 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.

As indicated in Section 4.0 above [of the application dated September 15, 2000,] ASME Section III stress limits for effected components are not exceeded. The evaluations indicate that the reactor vessel will continue to meet ASME Code allowable stress criteria with a single untensioned reactor vessel closure stud, or with a single closure stud which fails in service. The proposed change does not alter nor exceed the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

Therefore, the proposed change to the Technical Specifications do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration. The licensee's reference to ''closure studs'' in the above not significant hazards consideration is a reference to the ''closure bolts'' in the proposed amendment.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: September 7, 2000.

Description of amendment request: The amendments revise Surveillance Requirement 3.8.1.9.a by adding a note that states the upper limits on frequency and voltage are not required to be met for the annual test of the Keowee Hydro Units until the NRC issues an amendment that removes the note in response to an amendment request to be submitted no later than April 5, 2001.

*Date of publication of individual notice in **Federal Register:*** September 19, 2000 (65 FR 56600).

Expiration date of individual notice: October 3, 2000, for comments; October 19, 2000, for hearings.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York

Date of application for amendment: July 14, 2000.

Brief description of amendment request: The proposed amendment would revise License Condition 2.c.(10), ''Additional Condition 1,'' which was imposed by Amendment No. 91 dated February 15, 2000. License Condition 2.c.(10) defines the meaning of implementation of Improved Technical Specifications and specifies that implementation be completed by August 31, 2000. The licensee has proposed to revise the implementation date from August 31, 2000, to December 31, 2000.

*Date of publication of individual notice in **Federal Register:*** July 27, 2000 (65 FR 46183).

Expiration date of individual notice: August 28, 2000.

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

Date of application for amendments: June 7, as supplemented June 23, and August 24, 2000.

Brief description of amendment: Changes to Facility Operating License and Technical Specifications to reflect an increase in allowable thermal power from 3411 to 3459 megawatts.

*Date of publication of individual notice in the **Federal Register:*** September 7, 2000 (65 FR 54322).

Expiration date of individual notice: October 10, 2000.

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 NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

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

Date of application for amendment: November 30, 1999.

Brief description of amendment: This amendment revises the testing requirements in Technical Specification 5.5.11, ''Ventilation Filter Testing Program (VFTP),'' in response to Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of issuance: September 14, 2000.

Effective date: September 14, 2000.

Amendment No.: 189.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73087).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 7, 2000, as supplemented on August 23, 2000.

Brief description of amendment: This amendment revises the surveillance test intervals and allowed outage times for Engineered Safety Features Actuation System (ESFAS) instrumentation in Technical Specification (TS) 3/4.3.2. It also revises the reactor trip system instrumentation requirements in TS 3/4.3.1 associated with implementing the ESFAS relaxations.

Date of issuance: September 13, 2000.

Effective date: September 13, 2000.

Amendment No.: 101.

Facility Operating License No. NPF–63. Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 12, 2000 (65 FR 43044).

The supplemental submittal dated August 23, 2000, provided clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 23, 2000, as supplemented by letters dated June 19 and July 17, 2000.

Brief description of amendments: The amendments changed Technical Specification (TS) 3/4.6.K to revise the reactor pressure-temperature (P–T) limits; changed TSs 1.0 and 3/4.12.C to delete a special test exception that allowed the hydrostatic test to be performed above 212 degrees Fahrenheit while in Mode 4; and added a condition to the Unit 2 and 3 licenses to specify expiration dates for the P–T limits.

Date of issuance: September 19, 2000.

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

Amendment Nos.: 179 and 174.

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

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17911).

The June 19 and July 17, 2000, letters are within the scope of the original notice and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 19, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 3, 1999, as supplemented by letter dated February 25, 2000.

Brief description of amendments: The amendments revised Technical Specification (TS) Section 2.1.B to reflect a change to the Minimum Critical Power Ratio for Unit 2; added an approved analytical method to TS Section 6.9.A.6 for Units 2 and 3 for use in determining core operating limits; and added conditions to the Unit 2 and 3 licenses to limit the maximum rod average burnup for any rod to 60 GWD/MTU until the staff has completed an environmental assessment supporting a greater limit.

Date of issuance: September 21, 2000.

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

Amendment Nos.: 180 and 175.

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

Date of initial notice in ***Federal Register:*** September 8, 1999 (64 FR 48859). The February 25, 2000, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in an Environmental Assessment dated September 19, 2000, and a Safety Evaluation dated September 21, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 29, 2000, as supplemented by letter dated July 5, 2000.

Brief description of amendments: The amendments revised the Technical Specifications Table 3.3.2–1, Engineered Safety Feature Actuation System Instrumentation, Function 6.f, Auxiliary Feedwater Pump Suction Pressure-Lo.

Date of issuance: September 13, 2000.

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

Amendment Nos.: 193 and 174.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51350).

The supplement dated July 5, 2000, provided clarifying information that did not change the scope of the February 29, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 13, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 6, 2000, as supplemented by letter dated July 20, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 3.3.1, ''Reactor Trip System Instrumentation;'' TS 3.3.2, ''Engineered Safety Features Actuation System Instrumentation;'' TS 3.3.5, ''Loss of Power Diesel Generator Start Instrumentation;'' and TS 3.3.6, ''Containment Purge and Exhaust Isolation Instrumentation.''

Date of issuance: September 18, 2000.

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

Amendment Nos.: 194/175.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15378).

The supplement dated July 20, 2000, provided clarifying information that did not change the scope of the January 6, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 18, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 29, 2000, as supplemented by letters dated July 27, and August 10, 2000. Other related information was submitted by letters dated April 10, April 17, and June 19, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to reference the Westinghouse Best Estimate Large Break Loss-of-Coolant Accident analysis methodology described in WCAP–12945–P–A, March 1998. The changes also address corresponding TS Bases changes.

Date of issuance: September 22, 2000.

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

Amendment Nos.: 195 and 176.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51349).

The supplements dated April 10, April 17, June 19, July 27, and August 10, 2000, provided clarifying information that did not change the scope of the June 29, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 22, 2000.

No significant hazards consideration comments received: No.

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: April 26, 1999; supplemented May 15, July 26, and August 23, 2000.

Brief description of amendments: The amendments revised various provisions of the Technical Specifications and Final Safety Analysis Report related to the steam generator tube loads following a main steam line break and runout protection for the turbine-driven emergency feedwater pump.

Date of Issuance: September 18, 2000.

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

Amendment Nos.: 315, 315, & 315.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications and Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** May 19, 1999 (64 FR 27320).

The supplements dated May 15, July 26, and August 23, 2000, provided clarifying information that did not change the scope of the April 26, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 18, 2000.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of application for amendment: April 13, 2000, as supplemented by letter dated May 15, 2000.

Brief description of amendment: The amendment revised Surveillance Requirements 3.3.1.1.10 for Function 8 of Table 3.3.1.1–1 and 3.3.4.1.2.a. for reactor protection system and end of cycle recirculation pump trip instrumentation of the WNP–2 technical specifications. The amendment extends the frequency of these surveillance requirements from 18 months to 24 months.

Date of issuance: September 15, 2000.

Effective date: September 15, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 168.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37423).

The May 15, 2000, supplemental letter provided clarifying information, did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 15, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: July 28, 1999, as supplemented on June 6, 2000.

Brief description of amendment: This amendment revised the ultimate heat sink (UHS) average water temperature from 85 degrees Fahrenheit (°F) to ≤90 °F and permits plant operations in Operating Modes 1 through 4 with an average water temperature of ≤90 °F.

Date of issuance: September 12, 2000.

Effective date: Immediately, to be implemented within 90 days.

Amendment No.: 242.

Facility Operating License No. NPF–3: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 25, 1999 (64 FR 46438). The June 6, 2000, submittal provided additional 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 September 12, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: September 7, 1999, supplemented July 14, 2000.

Brief description of amendment: This amendment revises Technical Specification 3/4.3.2.1, Safety Features Actuation System Instrumentation Trip Setpoints, to remove the ''Trip Setpoint'' values for Instrument String Functional Unit ''b'', Containment Pressure-High, and Functional Unit ''c'', Containment Pressure-High-High, and also modifies the ''Allowable Values'' entry for these same Functional Units, consistent with updated calculations using current setpoint methodology. The changes also revise Limiting Conditions for Operation (LCO) 3.3.2.1, and Bases 3/4.3.1 and 3/4.3.2 to reflect the removal of the ''Trip Setpoint'' values for these Functional Units.

Date of issuance: September 14, 2000.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment No.: 243.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70086).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of application for amendment: August 4, 1999, and as supplemented by letter dated August 7, 2000.

Brief description of amendment: This amendment revised Technical Specification 3.9.1, ''Refueling Equipment Interlocks,'' by introducing an optional operator action when one or more required refueling equipment interlocks are inoperable. The new operator action permits continued in-vessel fuel movement under specific administrative controls.

Date of issuance: September 12, 2000.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment No.: 116.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 25, 1999 (64 FR 46439).

The August 7, 2000, supplement contained clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 12, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 16, 1999, as supplemented by letters dated May 3 and June 29, 2000.

Brief description of amendment: To allow an increase in the spent fuel pool (SFP) storage capacity by replacing fuel racks in the ''B'' SFP with new high-density fuel racks.

Date of issuance: September 13, 2000.

Effective date: September 13, 2000.

Amendment No.: 193.

Facility Operating License No. DPR–31: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 8, 1999 (64 FR 68702).

The May 3 and June 29, 2000 supplements provided clarifying information and did not affect the initial no significant hazards determination.

The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated September 5, 2000 and in a Safety Evaluation dated September 13, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 18, 1999, as supplemented on June 22 and December 10, 1999, and February 10, and May 2, 2000.

Brief description of amendment: The amendment revised the Technical Specifications to reflect the installation of additional spent fuel pool storage racks. The additional new racks will provide 390 additional spent fuel assembly storage locations.

Date of issuance: September 15, 2000.

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

Amendment No.: 215

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

*Date of initial notice in **Federal Register:*** August 17, 1999 (64 FR 44757).

The supplemental letters dated June 22 and December 10, 1999, and February 10 and May 2, 2000, did not affect the proposed finding of no significant hazards consideration, and was within the scope of the amendment application as noticed.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 15, 2000.

No significant hazards consideration comments received: No.

North Atlantic Energy Service Corporation, et al., Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: February 18, 2000, as superseded by a letter dated June 20, 2000.

Description of amendment request: The amendment changes the Seabrook Station Technical Specifications to provide operational flexibility during the shutdown modes of operation. These enhancements include: (1) The ability to have a standby Safety Injection (SI) pump available during Reactor Coolant System (RCS) reduced inventory conditions with the RCS pressure boundary intact; (2) realigning a footnote to clarify the allowance of an inoperable SI pump to be energized for testing or filling accumulators; (3) allowance for an additional charging pump to be made capable of injection during pump-swap operations; (4) recognition that a substantial vent area exists for cold overpressure protection when the reactor vessel head is on and the studs are fully detensioned; (5) limit maneuvering the plant beyond Hot Shutdown when one charging pump is operable; and (6) establishes a new value for the open permissive interlock associated with the Residual Heat Removal System suction isolation valves.

Date of issuance: September 11, 2000.

Effective date: As of its date of issuance, and shall be implemented within 60 days.

Amendment No.: 74.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 9, 2000 (65 FR 48752).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 2000.

No significant hazards consideration comments received: No.

North Atlantic Energy Service Corporation, et al., Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: December 3, 1999.

Description of amendment request: The amendment changes the Technical Specifications by incorporating reference to the American Society for Testing and Materials (ASTM) Standard D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Charcoal,'' as the test protocol for charcoal filter laboratory testing. In addition, there is a change to Surveillance Requirement 4.7.6.1d.5) and 4.9.12d.4) specifying a minimum required heater output based on design rated voltage.

Date of issuance: September 19, 2000.

Effective date: As of its date of issuance, and shall be implemented within 60 days.

Amendment No.: 75

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4282).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 19, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 1, 2000, as supplemented on June 1 and July 13, 2000.

Brief description of amendment: The amendment modifies Technical Specification (TS) 3.0.3 to state that this specification is not applicable in MODES 5 or 6. The amendment also makes various changes to TSs 3/4.1 ''Reactor Coolant System—Coolant Loops and Coolant Recirculation'' and 3/4.9.8, ''Refueling Operations—Shutdown Cooling and Coolant Circulation.'' In addition, various corrections and formats are revised to achieve consistency of the structure and wording of the TSs. The Bases for the affected TSs have also been revised accordingly.

Date of issuance: September 14, 2000.

Effective date: As of the date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment No.: 249.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 31, 2000 (65 FR 46748).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 2000.

No significant hazards consideration comments received: No.

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 18, 1999, and supplemented by letters dated April 5 and December 21, 1999; and May 2 and August 10, 2000.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) and the Final Safety Analysis Report for Millstone 3 to allow an entire reactor core to be offloaded to the spent fuel pool (SFP) and an increase in the maximum design basis normal SFP water temperature limit from 140 °F to 150 °F during planned refueling outages. The increase in maximum design basis normal SFP water temperature up to 150 °F affects certain Fuel Building area TS temperature limits that require a revision to the TSs.

Date of issuance: September 12, 2000.

Effective date: As of the date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment No.: 182.

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

Date of initial notice in ***Federal Register:*** March 10, 1999 (64 FR 11962).

The letters dated April 5 and December 21, 1999, and May 2, and August 10, 2000, provided clarifying information and did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the application as published in the **Federal Register**.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 12, 2000.

No significant hazards consideration comments received: No.

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: February 3, 2000

Brief description of amendment: The amendment changes the Millstone Unit No. 3 Final Safety Analysis Report to show that the configuration of valves 3CHS*V61 and 3CHS*V62 takes exception to the American Society of Mechanical Engineers Section III code requirements for class 2 components.

Date of issuance: September 15, 2000.

Effective date: As of the date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment No.: 183.

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

Date of initial notice in ***Federal Register:*** June 28, 2000 (65 FR 39958).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 15, 2000.

No significant hazards consideration comments received: No.

PECO Energy Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, (LGS) Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: November 5, 1999, as supplemented July 17, 2000.

Description of amendment request: The amendments make changes to Technical Specifications Sections 4.6.5.3.b.2 and 4.6.5.3.c, ''Standby Gas Treatment System,'' 4.6.5.4.b.2 and 4.6.5.4.c, ''Reactor Enclosure Recirculation System,'' and 4.7.2.c.2 and 4.7.2.d, ''Control Room Emergency Air System.''

Date of Issuance: September 8, 2000.

Effective Date: September 8, 2000.

Amendment Nos.: 144 & 106.

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

Date of Initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4287)

The July 17, 2000, letter provided clarifying information that did not change the initial no significant hazards consideration determination or expand the scope of the original **Federal Register** Notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 8, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 15, 2000, as supplemented August 10, 2000.

Brief description of amendment: The amendment revises the pressure-temperature limit curves.

Date of issuance: September 15, 2000.

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

Amendment No.: 145.

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

Date of initial notice in ***Federal Register:*** July 12, 2000 (65 FR 43051).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 15, 2000. The August 10, 2000, letter provided clarifying information that did not change the initial no significant hazard consideration determination or expand the scope of the original **Federal Register** notice.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 20, 2000 (PCN–503), and supplemented by letter dated June 6, 2000.

Brief description of amendments: The amendments revise TS 5.5.2.5, ''Reactor Coolant Pump Flywheel Inspection Program'' by changing the volumetric examination frequency of the upper flywheel on each of the primary reactor coolant pump motors from a 3-year to a 10-year cycle.

Date of issuance: September 8, 2000.

Effective date: September 8, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—170; Unit 3—161.

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

Date of initial notice in ***Federal Register:*** May 17, 2000 (65 FR 31360). The supplemental letter dated June 6, 2000, provided clarifying information that was within the scope of the April 20, 2000, application and the **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 8, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 12, 2000.

Brief description of amendments: The amendments revise TS 3.6.6.1, ''Containment Spray and Cooling Systems,'' to change the allowed outage time (AOT) for a single inoperable train of the containment spray system from 72 hours to 7 days. Also, the combined AOT that appears in both Conditions A and C of TS 3.6.6.1 is revised from 10 days to 14 days.

Date of issuance: September 12, 2000.

Effective date: September 12, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—171; Unit 3—162.

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

Date of initial notice in ***Federal Register:*** May 3, 2000 (65 FR 25769).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 12, 2000.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: March 6, 2000, as supplemented by letter dated July 7, 2000.

Brief description of amendments: The amendments revised Technical Specification (TS) 3.9.4, ''Containment Penetration,'' allowing the equipment hatch to be open during core alteration and/or during movement of irradiated fuel within the containment, provided the capability for closure is maintained.

Date of issuance: September 11, 2000.

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

Amendment Nos.: 115 and 93.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 28, 2000 (65 FR 39961), July 20, 2000 (65 FR 45115). The supplemental letter dated July 7, 2000, provided clarifying information that did not change the scope of the March 6, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 11, 2000.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–3a0, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: July 10, 2000 (TS 00–08).

Brief description of amendment: Regarding the need to conduct channel operational tests within 12 hours prior to physics tests and the placing of a reactor trip instrumentation channel used in physics tests in a bypassed condition instead of a tripped condition.

Date of issuance: September 13, 2000.

Effective date: September 13, 2000.

Amendment No.: 28.

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

Date of initial notice in ***Federal Register:*** August 9, 2000 (65 FR 48759).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 23, 2000, and supplements dated July 21 and 26, 2000.

Brief description of amendment: The amendment revises TS 3.9.4, ''Containment Penetrations,'' to allow containment penetrations (with direct access to the outside atmosphere) to be unisolated under administrative controls during refueling operations with core alterations or irradiated fuel movement inside containment. The amendment (1) revises the note in the Limiting Condition for Operation 3.4.9 for containment penetrations that may be unisolated under administrative controls, deleting the reference to penetrations P–63 and P–98, and (2) deletes the exception for penetrations P–63 and P–98 in Surveillance Requirement 3.9.4.1. In addition, there are format and editorial corrections to TS 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Start Air,'' and TS 5.2.2.b, ''Administrative Controls,'' to correct errors issued in Amendment No. 123, issued March 31, 1999.

Date of issuance: September 12, 2000.

Effective date: September 12, 2000, to be implemented within 30 days of the date of issuance, including the completion of the administrative procedures that ensure that open containment penetrations, with direct access to the outside atmosphere during refueling operations with core alterations and irradiated fuel movement inside containment, will be promptly closed in the event of a fuel handling accident inside containment.

Amendment No.: 135.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 12, 2000 (65 FR 43053). The July 21 and 26, 2000, supplements provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 12, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 28th day of September 2000.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Acting Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 00–25377 Filed 10–3–00; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Excepted Service

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: This gives notice of positions placed or revoked under Schedules A and B in the excepted service, as required by Civil Service Rule VI, Exceptions from the Competitive Service.

FOR FURTHER INFORMATION CONTACT: Janice Reid, Staffing Policy Division, Employment Service (202) 606–0830.

SUPPLEMENTARY INFORMATION: The Office of Personnel Management publishes this monthly notice to update appointing authorities established or revoked under the Excepted Service provisions of 5 CFR 213. Individual authorities established or revoked under Schedules A and B between August 1, 2000, and August 31, 2000, appear in the following listing. A consolidated listing of all authorities as of June 30 is published annually.

Schedule A

No Schedule A authorities were established or revoked during August 2000.

Schedule B

The following Schedule B authority was amended effective August 17, 2000.

Schedule B 213.3209

''(a) Not to exceed six interdisciplinary positions for the Airpower Research Institute at the Air University, Maxwell Air Force Base, Alabama, for employment to complete studies proposed by candidates and acceptable to the Air Force. Initial appointments are made not to exceed 3 years, with an option to renew or extend the appointments in increments of 1, 2, or 3 years indefinitely thereafter.''

Authority: 5 U.S.C. 3301 and 3302; E.O. 10577, 3 CFR 1954–1958 Comp., P.218.