AUTHENTICATED U.S. GOVERNMENT INFORMATION GPO

advised of any potential changes in the proposed agenda, etc., that may have occurred.

Dated: April 15, 1999.

Richard P. Savio,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 99–9937 Filed 4–20–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of April 19, 26, May 3 and 10, 1999.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of April 19

There are no meetings scheduled for the Week of April 19.

Week of April 26—Tentative

Monday, April 26

2:00 p.m.—Affirmation Section (Public Meeting) (if needed)

Week of May 3—Tentative

Tuesday, May 4

9:00 a.m.—Meeting on NRC Response to Stakeholders' Concerns (Public Meeting) Location: (NRC Auditorium, Two White Flint North)

2:00 p.m.—Meeting on Planning, Budgeting and Performance Management Process (PBPM) And Institutionalizing Change (Public Meeting)

Wednesday, May 5

9:00 a.m.—Discussion of Intragovernmental Issues (Closed-Ex. 9b)

10:00 a.m.—Briefing on Safeguards Performance Assessment (Public Meeting)

Thursday, May 6

9:30 a.m.—Briefing on Operating Reactors and Fuel Facilities (Public Meeting) (Contact: Glenn Tracy, 301–415–1725)

11:30 a.m.—Affirmation Session (Public Meeting) (if needed)

Week of May 10—Tentative

There are no meetings scheduled for the Week of May 10.

*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.

ADDITIONAL INFORMATION: By a vote of 5–0 on April 15, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of (a) Private Fuel Storage, LLC (PFS) Review of Board's Decision Granting Late-Filed Intervention Petition of Southern Utah Wilderness Alliance (LBP–99–3) (February 3, 1999) and (b) Duke Energy Corporation—Commission Review of LBP 98–33'' (PUBLIC MEETING) be held on April 15, and on less than one week's notice to the public.

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: April 16, 1999.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 99–10124 Filed 4–19–99; 8:45 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 27 through April 9, 1999. The last biweekly notice was published on April 7, 1999 (64 FR 17021).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration Services, Office of

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

By May 21, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: March 23, 1999.

Description of amendment request: The proposed amendment would modify Technical Specification Surveillance Requirement 4.4.1.1.1 to require each recirculation pump discharge valve to be demonstrated OPERABLE at least once every 18 months and will delete footnote * that applies to Technical Specification 4.4.1.1.1.

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

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

The proposed changes to the Technical Specifications (TS) would modify the frequency of cycling the recirculation pump discharge valves from ''each STARTUP* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER'' to ''at least once per 18 months;'' and replace the footnote applicable to TS 4.4.1.1, ''*If not performed in the previous 31 days'' with ''*Not Used.'' The change in testing frequency does not affect the probability of an accident since the valve testing is not related to accident initiation sequences. Consequences of accidents are not significantly increased because the proposed testing interval provides reasonable assurance that the valves will function. Testing of the valves will still be performed on a frequency that is allowed by TS if no events occur that require entry into Mode 3 or Mode 4. Therefore, the change will not involve a significant increase in the probability or consequences of an accident previously evaluated. Testing the valves in accordance with the inservice testing (IST) program on the same testing frequency as testing performed for the low pressure coolant injection system, provides adequate assurance that the valves can perform their safety function and will not increase the consequences of an accident previously evaluated. The change to the footnote is administrative in nature and will have no effect on the probability of an accident and will not increase any safety consequences.

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

The proposed changes revise performing the testing of the recirculation pump discharge valves from ''prior to Startup* not to exceed 25% of rated thermal power.'' to ''at least once per 18 months'' and replace the footnote applicable to TS 4.4.1.1'' *If not performed in the previous 31 days'' with ''*Not Used'' does not result in a new accident precursor since the test only verifies that the valve can close which is its safety function. Deleting the information contained in footnote ''*'' that applies to TS 4.4.1.1.1 and designating it as ''* Not Used.'' is administrative in nature with no safety significance. Therefore, no different type of accident from any previously evaluated is introduced.

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

The proposed changes revise the frequency of cycling the recirculation pump discharge valves from ''each STARTUP* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER'' to ''at least once per 18 months'' and replace the footnote applicable to TS 4.4.1.1 ''*If not performed in the previous 31 days'' with ''*Not Used.'' Altering the test frequency does not change valve stroke time or other performance or design characteristics related to the safety function of the valves. The potential for failure of the valve to close is not changed as a result of the proposed change since the same frequency is allowed by the current TS if no events occur that require entry into Mode 3 or Mode 4. Performing stroke time testing on a refueling outage basis and MOV testing on a periodic basis does not decrease the margin of safety associated with the valve performing its safety function. Revising footnote * is an administrative change and has no safety consequence. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Section Chief: George F. Dick, Acting.

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: March 3, 1999.

Description of amendment request: The proposed amendments would change the required qualifications for operations management specified in the Technical Specifications (TSs) for the Beaver Valley Power Station, Units 1 and 2 (BVPS–1 and BVPS–2). The requirement that the operations manager hold a Senior Reactor Operator (SRO) license at the time of appointment would be changed in the TSs to require that the assistant operations managers, one for each unit, hold an SRO license on their assigned unit. The TSs would not then require the operations manager hold an SRO license. Additionally, the Updated Final Safety Analysis Report (UFSAR) for each unit would be changed to require the operations manager to hold, or have held, an SRO license rather than presently hold a license. The UFSAR would require the same as the TS; that the assistant operations managers hold an SRO license on the unit to which they are assigned. Finally, the proposed amendments would substitute generic personnel titles for plant-specific personnel titles in the BVPS–1 and BVPS–2 TSs. The correlation between generic titles and plant-specific titles would be provided in the BVPS–2 UFSAR.

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

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

The proposed changes are administrative in nature. The revised requirements for who must hold a current senior reactor operator (SRO) License does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident nor alter the conditions or assumptions in any of the Updated Final Safety Analysis Report [UFSAR] accident analyses. The requirement that the operations manager hold or have held an SRO License is included in the revised Position Qualifications in the Unit 2 UFSAR, Table 13.1–2, sheet 30 of 35. The title changes are being made, consistent with TSTF–65, Rev 1 and help avoid the need for future Technical Specification changes. Therefore, it can be concluded that the proposed changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

No new failure modes are defined for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed changes. Therefore, it can be concluded that the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

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

The proposed changes are administrative in nature. One of the proposed changes requires that the manager who directly supervises the licensed operators at each unit be the holder of a current SRO license. The other change modifies personnel titles. Therefore, it can be concluded that the proposed changes do not involve any reduction in a margin of safety.

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

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

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

NRC Section Chief: Singh Bajwa.

Duquesne Light Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit No. 2, Shippingport, Pennsylvania

Date of amendment request: March 16, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.7.1.3

and associated Bases for the Primary Plant Demineralized Water (PPDW) System to clarify that the minimum specified volume of water in the PPDW Storage Tank is a usable volume. Additionally, the minimum usable volume of water in the PPDW Storage tank is increased, and a clarifying footnote that the specified value is an analysis value is added. Finally, several editorial and administrative changes, such as revision of action statement wording, addition of license number to the TS page, and addition of clarifying information to the TS Bases regarding analysis assumptions are made.

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

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

The failure of the primary plant demineralized water (PPDW) storage tank to provide a sufficient source of water to the Auxiliary Feedwater (AFW) System is not an accident initiating event. Therefore, the probability of an accident previously evaluated is not increased by this proposed amendment.

Limiting Condition for Operation (LCO) 3.7.1.3 titled ''Primary Plant Demineralized Water (PPDW)'' will be revised to specify the required value for PPDW storage tank volume as a usable volume. To reflect the value currently assumed in the analysis, the value stated in the LCO, for minimum required PPDW storage tank volume, would be slightly increased. The addition of proposed Footnote (1) to LCO 3.7.1.3 will ensure that plant operators recognize that the specified volume is an analysis value and that the value does not include measurement uncertainties. This footnote will require plant procedures to specify an increased required volume in the PPDW storage tank to account for measurement uncertainties. The proposed revisions to LCO 3.7.1.3 will assure that the PPDW storage tank minimum usable volume is maintained consistent with the design basis for the PPDW storage tank. The PPDW storage tank will continue to provide a sufficient source of water to the AFW pumps. Maintaining a sufficient source of water will ensure that the AFW System is capable of mitigating the consequences of Design Basis Accidents (DBAs) that could result in overpressurization of the RCS pressure boundary. The AFW system will continue to be capable of providing an emergency source of feedwater to the steam generators to act as heat sinks for sensible and decay heat removal from the reactor core. A sufficient volume of water will continue to be maintained in the PPDW storage tank to satisfy the Safe Shutdown evaluation.

The proposed changes to the Action statements will remove the required water volume value and add wording pertaining to the water volume not being within the limit. The LCO clearly states the value for the minimum required volume in the PPDW storage tank. Therefore, the proposed modification to the Action statements is administrative in nature and does not affect plant safety. The additional Bases wording pertaining to reactor coolant pump operation is administrative in nature and does not affect plant safety. The remaining change, which consists of the addition of plant operating license number, is editorial in nature and does not affect plant safety.

Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed amendment will not change the physical plant or the modes of plant operation defined in the operating license. This change does not involve the addition or modification of plant equipment nor does it alter the design or operation of plant systems. The proposed amendment will require that the minimum volume in the PPDW storage tank be maintained consistent with analysis assumptions.

Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The minimum required volume in the PPDW storage tank would be slightly increased over the currently required value. This increase in the required volume will ensure that an adequate volume of water is maintained in the PPDW storage tank. The proposed addition of the term ''usable,'' along with the addition of Footnote (1), will ensure that the water volume specified in LCO 3.7.1.3 is appropriately increased in plant procedures to account for unusable volume in the tank and for measurement uncertainties. A sufficient volume of water will continue to be maintained in the PPDW storage tank to satisfy the Safe Shutdown evaluation.

The PPDW storage tank will continue to provide a sufficient source of water to the AFW pumps to ensure that the AFW System is capable of mitigating the consequences of DBAs that could result in overpressurization of the RCS pressure boundary. The AFW system will continue to be capable of providing an emergency source of feedwater to the steam generators to act as heat sinks for sensible and decay heat removal from the reactor core.

The proposed changes to the Action statements will remove the required water volume value and add wording pertaining to the water volume not being within the limit. The LCO clearly states the value for the minimum required volume in the PPDW storage tank. Therefore, the proposed modification to the Action statements is administrative in nature and does not affect plant safety. The additional Bases wording pertaining to reactor coolant pump operation is administrative in nature and does not affect plant safety. The remaining change, which consists of the addition of plant operating license number, is editorial in nature and does not affect plant safety.

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

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

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

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

NRC Section Chief: S. Singh Bajwa.

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, 1998, and revised March 18, 1999.

Description of amendment request: The proposed amendment would revise Improved Technical Specification (ITS) 5.6.2.10, ''Steam Generator (OTSG [once-through steam generator]) Tube Surveillance Program,'' to include a new repair process, called a ''repair roll'' or ''re-roll.'' The process would be used to repair steam generator tubes with defects within the upper tubesheet. Changes to inservice inspection and reporting requirements are proposed for tubes which are repaired using this process. In addition, several format and editorial changes are proposed to ITS 5.6.2.10 and to ITS 5.7.2, ''Special Reports,'' for clarification purposes. The March 18, 1999 revision superceded the August 31, 1998 request, and includes the results of recent accident analyses conducted to identify the maximum OTSG tube tensile loads. As a result of the increased tube tensile loads, some tubes will require a double repair roll. The double repair roll methodology was not included in the original amendment request. Therefore, this notice revises the previous Notice of Consideration of Issuance of Amendment (63 FR 56249).

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

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

The repair roll process is a method to create a new primary-to-secondary pressure boundary joint in the upper tubesheet of Babcock & Wilcox (B&W) Once Through Steam Generators (OTSGs) manufactured with Inconel Alloy 600 tubes. The repair roll process creates a new roll joint in the OTSG tubes at a point closer to the secondary face of the tubesheet than the existing roll joint. The new pressure boundary is established by the repair roll to remove degradation of the existing roll joint from pressure boundary service. The repair roll process has been qualified as an acceptable repair methodology for use in the upper tubesheet of the Crystal River Unit 3 (CR–3) OTSGs. The proposed License Amendment Request (LAR) proposes to implement the qualified OTSG tube repair roll process, and also addresses several editorial and format changes which do not impact the current CR–3 accident analyses.

The qualification of the OTSG tube repair roll methodology is based on establishing a mechanical joint length that will carry all structural loads imposed on the OTSG tubes while maintaining the required margins during normal and accident conditions. A series of tests and analyses were performed to establish the minimum acceptable length of the OTSG tube repair roll. Tests performed included leak, tensile, fatigue, ultimate load and eddy-current measurement uncertainty. The analyses evaluated plant operating and faulted load conditions, in addition to OTSG tubesheet bow effects. OTSG tube leakage remains bounded by the evaluation presented in the CR–3 Final Safety Analysis Report (FSAR) for a main steam line break (MSLB). The proposed change also includes a description of the required inspection program for the OTSG tube repair rolls. The additional inspection requirements do not change any accident initiators. The proposed inspections following OTSG tube repair roll installation, and during future inservice inspections, assure continuous monitoring of these tubes such that inservice degradation of tubes repaired by the repair roll process will be detected. Based on the qualification testing and analyses performed, as well as the industry experience with the use of OTSG tube repair roll processes, there are no new safety issues associated with the use of repair roll methodology. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from previously evaluated accidents?

The repair roll creates no new failure modes or accident scenarios. The new pressure boundary joint created by the repair roll process has been demonstrated, by testing and analysis, to provide structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. Furthermore, the testing and analysis demonstrate the repair roll process creates no new adverse effects for the repaired tube and does not change the design or operating characteristics of the OTSGs. In the unlikely event that a tube with a repair roll should fail and sever completely at the transition of the repair roll region, the tube would remain engaged in the tubesheet bore, preventing interaction with other surrounding tubes. In this case, leakage is bounded by the steam generator tube rupture (SGTR) accident analysis. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Involve a significant reduction in a margin of safety?

The repair roll process effectively removes the defective/degraded area of the tube from service. The repair roll interface created with the tubesheet satisfies the necessary structural, leakage and heat transfer requirements. The mechanical joint is constrained within the tubesheet bore; thus, there is no additional risk associated with tube rupture. The accident leakage is shown to be less than one gallon per minute primary-to-secondary leakage. Therefore, the FSAR analyzed accident scenarios remain bounding, and the use of the repair roll process does not reduce the margin of safety.

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

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

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: Sheri R. Peterson.

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

Date of amendment request: March 1, 1999.

Description of amendment request: The proposed amendment would approve changes to the Updated Safety Analysis Report (USAR) concerning design requirements for physical protection from tornado missiles for safety-related equipment.

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 activity does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The associated USAR changes reflect use of the Electric Power Research Institute (EPRI) Topical Report, ''Tornado Missile Risk Evaluation Methodology, (EPRI NP–2005),'' Volumes I and II. This methodology has been reviewed, accepted and documented in an NRC Safety Evaluation dated October 26, 1983. The NRC concluded that: ''the EPRI methodology can be utilized when assessing the need for positive tornado missile protection for specific safety-related plant features in accordance with the criteria of SRP Section 3.5.1.4.''

The EPRI methodology has been previously applied at CPS to resolve previously identified missile protection issues during the initial licensing of the plant. The NRC documented their acceptance of this methodology in Supplement 6 to the CPS Safety Evaluation Report (NUREG–0853, July 1986).

As permitted in the Standard Review Plan (NUREG–0800), the total probability of damage to plant systems or components initiated from tornado missiles leading to consequences in excess of 10 CFR Part 100 guidelines will be maintained below an acceptable level. The results of the current tornado missile hazards analysis are such that the calculated total tornado missile hazard probability is approximately 3.4×10^{-7} per year. This is lower than the value determined to be acceptable, i.e., 1×10^{-6} per year.

Although it has been calculated that these targets have a higher total probability of being exposed to tornado missiles than that described to be acceptable in SER Supplement 6, Section 3.5.1.3, the revised tornado missile hazards analysis for CPS has determined that this probability is acceptably low.

With respect to the probability of occurrence or the consequences of an accident previously analyzed in the USAR, the possibility of a tornado reaching CPS and causing damage to plant systems, structures and components is a design basis event considered in the USAR. The changes being proposed herein do not affect the probability that a tornado will reach the plant, but they do, from a licensing basis perspective, reflect a slightly increased, calculated probability that missiles generated by the winds of a tornado might strike certain plant systems or components. The tornado missile analysis determined that there are a limited number of safety-related components that theoretically could be struck. The probability of tornado-generated missile strikes on important systems and components (as discussed in Regulatory Guide 1.117) was analyzed using the probability methods described above. Based on the low, calculated probability, the total (cumulative) probability of strikes will be maintained below an adequately low acceptance criterion to ensure overall plant safety. On this basis, the proposed change is not considered to constitute a significant increase in the probability of occurrence or the consequences of an accident, due to the low probability of a tornado missile striking safety-related systems or components.

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

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

The proposed changes involve evaluation of whether any physical protection of safety-related equipment from tornado missiles is required relative to the probability of such damage without physical protection. A tornado at CPS is a design basis event considered in the USAR, however, a tornado is not postulated to act as an initiator for any new or different kind of accident, or to occur coincident with any of the design basis accidents in the USAR. The low probability threshold established for missile damage to plant systems is consistent with these assumptions.

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

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

Under the proposed change, physical protection of safety-related equipment from tornado missiles must be considered if it has been determined that the calculated total tornado missile hazard probability is greater than 1×10^{-6} per year. The proposed change to the USAR to specifically identify this threshold may slightly increase the probability of a malfunction of equipment important to safety previously evaluated in the safety analysis report (i.e., changing the requirements from protecting all safety-related systems and components to not requiring protection if there is an extremely low probability that a tornado missile could strike portions of safety related systems and components). However, the changes are consistent with the minimum acceptable requirements as documented in the NRC's Safety Evaluation Report dated October 23, 1983. Therefore, there will be no significant reduction to the margin of safety that may be associated with the potential for safety-related equipment to be damaged from tornado-generated missiles.

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

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 17, 1999.

Description of amendment request: The licensee is proposing to change Technical Specifications 3.5.2, "Emergency Core Cooling Systems—ECCS Subsystems—Tavg greater than or equal to 300 °F;" 3.7.1.7, "Plant Systems—Atmospheric Steam Dump Valves;" and 3.7.6.1, "Plant Systems—Control Room Emergency Ventilation System." The proposed Technical Specification changes will revise (1) surveillance requirements for Emergency Core Cooling System valves, (2) the atmospheric steam dump valve requirements to focus on the steam release path instead of the individual valves, and (3) the allowed outage times for the atmospheric steam dump valves and Control Room Emergency Ventilation System.

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:

In accordance with 10 CFR 50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

Technical Specification 3.5.2

The removal of 2–CH–434, a manual valve, from the list of valves to be checked every 31 days by Surveillance Requirement (SR) 4.5.2.a.10 will not change the requirement for this containment isolation valve to be locked closed. The position of valve 2–CH–434, and the associated locking device, will be verified by SR 4.6.1.1.a. Although this change will result in the position of 2–CH–434 being checked less often, there are sufficient Technical Specification and administrative requirements to ensure that 2–CH–434 will be maintained in the proper position. An additional benefit of this proposed change will be a reduction in personnel exposure since 2–CH–434 is located inside containment. This proposed change will not result in any modification to Emergency Core Cooling System (ECCS) alignment or operation.

The addition of the footnote to SR 4.5.2.a.10 will clarify that 2–SI–306 is pinned and locked open to the required throttle position. 2–SI–306, which is the Shutdown Cooling (SDC) System throttle valve in the discharge piping of the SDC pumps, is required to be left in a throttled position after SDC has been secured to ensure sufficient low pressure safety injection (LPSI) flow will be available. This proposed change will not result in any modification to ECCS alignment or operation.

The change in the valve nomenclature used in SR 4.5.2.e and Table 4.5–1 from throttle valve to injection valve will eliminate any confusion between valve description and valve operation. This proposed change will not result in any modification to ECCS alignment or operation.

The addition of the License Amendment Number to the bottom of Page 3/4 5–6a will not result in a technical change to this Technical Specification.

Technical Specification 3.7.1.7

The proposed changes will expand the scope of Technical Specification 3.7.1.7 to include the steam release path, instead of just the individual atmospheric dump valves (ADVs). The allowed outage times will be modified to address inoperable ADV lines and the impact inoperable ADV lines will have on the ability of Millstone Unit No. 2 to mitigate a loss of coolant accident (LOCA). If one ADV line is inoperable, a plant shutdown will be required if the ADV line is not restored to operable status within 48 hours. An allowed outage time of 48 hours to restore the ADV line to operable status is acceptable based on the low probability of a LOCA occurring during this time period, and the subsequent loss of offsite power and the failure of one train of high pressure safety injection (HPSI). This is also consistent with the allowed outage time for one ECCS train (Technical Specification 3.5.2).

If two ADV lines are inoperable, a plant shutdown will be required if at least one ADV line is not restored to operable status within one hour. The plant will be required to be in Mode 3 within the following 6 hours. These time requirements are based on Technical Specification 3.0.3. However, the time to reach Mode 4 will remain at the "following 24 hours" to reflect the impact inoperable ADV lines may have on the time to cool down the plant.

The proposed change to the surveillance requirement will ensure operation of the ADV lines, consistent with the accident analysis, is verified.

The proposed change in component nomenclature is consistent with current Millstone Unit No. 2 terminology. This is not a technical change.

The proposed changes to the Bases of Technical Specification 3.7.1.7 are consistent with the changes just described.

Technical Specification 3.7.6.1

The action requirements for the Control Room Emergency Ventilation System will be modified to address the situation when both Control Room Emergency Ventilation Trains are inoperable in Modes 1, 2, 3, and 4. This situation is expected to occur during normal plant operation when the air filters in the common supply header to both trains are cleaned/replaced. Since this is a common supply header, both trains are affected and would be inoperable. The proposed action requirements will address this situation so that Technical Specification 3.0.3 will not be entered as a result of an expected plant activity. However, since the proposed action requirements are the same as the requirements of Technical Specification 3.0.3, the time the plant is allowed to operate in this situation will not change.

The proposed changes to the Technical Specifications and associated Bases will have no adverse effect on plant operation or accident mitigation equipment. The proposed changes will ensure that the necessary equipment to mitigate the design basis accidents will be available, or a plant shutdown will be required. In addition, the proposed changes can not cause an accident, and they will ensure the accident mitigation equipment will continue to operate as assumed in the analyses to mitigate the design basis accidents. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the Technical Specifications and associated Bases will have no adverse effect on plant operation or accident mitigation equipment. The proposed changes will ensure that the necessary equipment to mitigate the design basis accidents will be available, or a plant shutdown will be required. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to the Technical Specifications and associated Bases will ensure that the necessary equipment to mitigate the design basis accidents will be available, or a plant shutdown will be required. The proposed changes will not result in any plant configuration changes. There will be no adverse effect on plant operation or accident mitigation equipment. The plant response to the design basis accidents will not change. Therefore, there will be no significant reduction in the margin of safety as defined in the Bases for the Technical Specifications affected by these proposed changes.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: March 19, 1999.

Description of amendment request: The proposed changes will relocate Technical Specifications (TSs) 3.3.3.2, ''Instrumentation, Incore Detectors,'' 3.3.3.3, ''Instrumentation, Meteorological Instrumentation,'' to the Millstone, Unit No. 2 Technical Review Manual (TRM). Index Page V will be revised by eliminating the sections corresponding to incore detectors (Page ¾ 3–0), seismic instrumentation (Page ¾ 3–32), and meteorological instrumentation (Page ¾ 3–36). These sections, as well as changes to the associated Bases, will be relocated to the TRM.

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:

In accordance with 10 CFR 50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

Technical Specification 3.3.3.2, Instrumentation, ''Incore Detectors,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. Relocation of this Technical Specification to the TRM does not imply any reduction in its importance in confirming that core power distribution are bounded by safety analysis limits. These instruments are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary before a design basis accident, nor do they function as a primary success path to mitigate events which assume a failure of, or a challenge to, the integrity of fission product barriers. Although the core power distribution (measured by the incore detectors) constitutes an important initial condition to design basis accidents and therefore needs to be addressed by Technical Specifications, the detectors themselves are not an active design feature needed to preclude analyzed accidents or transients. The proposed change will not alter the way core power distribution is measured by the incore detectors, nor will it alter any of the power distribution assumptions used in the accident analysis. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3.3.3.3, Instrumentation, ''Seismic Instrumentation,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. Relocation of Technical Specification 3.3.3.3 to the TRM does not imply any reduction in its importance in determining the response of those nuclear power plant features important to safety in the event of an earthquake. Seismic instrumentation does not actuate any protective equipment or serve any direct role in the mitigation of an accident. The capability of the plant to withstand a seismic event or other design basis accident is determined by the initial design and construction of systems, structures, and components. The instrumentation is used to alert operators to the seismic event and evaluate the plant response. The seismic instrumentation does not serve as a protective design feature or part of a primary success path for events which challenge fission product barriers. The proposed change will not alter the way these instruments are used in determining the response of those nuclear power plant features important to safety in the event of an earthquake, nor will it alter the capability of the plant to withstand a seismic event. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3.3.3.4, Instrumentation, ''Meteorological Instrumentation,'' is proposed to be relocated to the TRM where future changes will be controlled in accordance with 10 CFR 50.59. Relocation of Technical Specification 3.3.3.4 to the TRM does not imply any reduction in its importance in providing a basis for estimating annual radiation doses resulting from radioactive materials released in airborne effluents. The instrumentation does not serve to ensure that the plant is operated within the bounds of initial conditions assumed in design basis accident and transient analyses or that the plant will be operated to preclude transients or accidents. Likewise, the meteorological instrumentation does not serve as part of the primary success path of a safety sequence analysis used to demonstrate that the consequences of these events are within the appropriate acceptance criteria. The proposed change will not alter the way these instruments are used in providing a basis for estimating annual radiation doses resulting from radioactive materials released in airborne effluents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Revision of Index page V and the proposed changes to the associated Bases sections are administrative changes. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes do not alter how any structure, system, or component functions. There will be no effect on equipment important to safety. The proposed changes have no effect on any of the design basis accidents previously evaluated. Therefore, this License Amendment Request does not impact the probability of an accident previously evaluated, nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

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

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

The proposed relocation of incore detector instrumentation requirements to the TRM does not imply any reduction in their importance in confirming that core power distribution is bounded by safety analysis limits. The incore detectors will still be used to measure core power distribution and the assumptions used in the accident analysis will be verified. The proposed relocation of seismic instrumentation requirements to the TRM does not imply any reduction in their importance in determining the response of those nuclear power plant features important to safety in the event of an earthquake. The seismic instrumentation will still be used to determine the response of those nuclear power plant features important to safety in the event of an earthquake. The capability of the plant to withstand a seismic or other design basis accident, which is determined by the initial design and construction of systems, structures, and components will not be altered. The relocation of meteorological instrumentation requirements to the TRM does not change the way these instruments are used in providing a basis for estimating annual radiation doses resulting from radioactive materials released in airborne effluents. The meteorological instrumentation will continue to perform their function in exactly the same way.

The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, the proposed changes will not result in a significant reduction in the margin of safety as defined in the Bases for Technical Specifications covered in this License Amendment Request.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 11, 1998, as supplemented by letter dated January 14, 1999.

Description of amendment request: The proposed amendments would change the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise TS 6.8.4f., ''Containment Polar and Turbine Building Cranes,'' to control the operation of the containment polar cranes in jet impingement zones.

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 Technical Specification (TS) 6.8.4f requirement to have a program that will ensure the position of the polar cranes precludes jet impingement from a postulated pipe rupture was previously evaluated in the NRC staff's safety evaluation for License Amendments (LA) 20 and 21. The proposed change is to control the operation of the containment polar cranes in jet impingement zones.

PG&E evaluated a high energy line break (HELB) scenario for core damage frequency (CDF) considering operation of a polar crane. A postulated HELB would have to damage the crane or cause its load to drop in a manner that damages a component that exacerbates the HELB event and leads to core damage. The PRA evaluation for this scenario concluded the CDF is 1.6E–9 per year. It is not a significant increase in CDF compared to never operating the polar crane in jet impingement zones. The CDF for this scenario is nonrisk significant when compared to the industry standard threshold for risk significance for an operational evolution, which is 1E–6 per year. Several factors that further lower the risk of CDF include: 1) the movement of heavy loads is done in accordance with the DCPP Heavy Loads Program, which provides assurance that a dropped load would not lead to core damage, 2) the polar crane had been evaluated to withstand jet impingement loads without the seismic loads, and 3) the probability of simultaneous seismic and HELB events is low.

Therefore, based on probabilistic considerations, the risk associated with this proposed change does 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.

Deterministic engineering methods required combining both the seismic and jet impingement loads to qualify Design Class I structures. The polar cranes were not originally qualified for these combined loads. This resulted in administrative controls that prohibited parking the polar cranes in jet impingement zones to preclude jet impingement loads from a postulated pipe rupture. The proposed change does not involve a physical change to the plant, but it does involve a change to the TS required program for containment polar crane operation.

The proposed change is to control the operation of the containment polar cranes in jet impingement zones. It recognizes that there are jet (HELB) and target (polar crane) interactions. They were previously not considered for postulated jet impingement analyses because administrative controls prohibited parking the polar cranes in jet impingement zones. PG&E has evaluated jet impingement loads on the polar crane and determined it is able to withstand these loads without seismic loads. Based on this evaluation, the polar crane would not fail due to a HELB event. The movement of a heavy load would be done in accordance with the DCPP Heavy Loads Program. Thus, there would be no consequential failures that would lead to core damage.

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

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

The current TS 6.8.4f. requirement to have a program that will ensure the position of the polar cranes precludes jet impingement from a postulated pipe rupture was previously evaluated in the NRC staff's safety evaluation for LAs 20 and 21.

The credible HELB sources that could impinge on the polar crane were identified and evaluated. The feedwater and main steam line steam generator nozzles are the only credible HELBs that could impinge upon the polar crane. The structural integrity of these lines was evaluated and determined to be of robust design.

The margin of safety affected by the proposed change involves a comparison between the margin of safety afforded by no operation of the polar crane and operation that is controlled by procedures. The margin of safety in this case is the increase in risk for CDF caused by a scenario that postulates that operation of the polar crane would lead to core damage. The risk for CDF has been evaluated and determined to be nonrisk significant. The CDF value is well below the industry standard threshold for acceptable risk for an operational evolution, which is 1E–6 per year.

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

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

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

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: December 12, 1998.

Description of amendment request: The proposed amendments would change the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise TS 6.9.1.8, ''Core Operating Limits Report,'' to allow use of NRC approved addenda to WCAP–10054–P–A, ''Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code,'' August 1985, to determine core operating limits.

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

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

This change is administrative in nature in that it revises the Technical Specification (TS) Administrative Controls for the Core Operating Limits Report to include reference to NRC approved addenda to WCAP–10054–P–A, ''Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,'' August 1985. The proposed change would allow the use of the analytical methods in WCAP–10054–P–A, Addendum 2, Revision 1, Addendum to the Westinghouse Small Break ECCS.

Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model,'' July 1997, and other NRC approved addenda to WCAP–10054–P–A to determine core operating limits for Diablo Canyon Power Plant (DCPP). Because plant operation will continue to be limited in accordance with cycle specific core operating limits that are established using an NRC approved methodology, NRC approved addenda to WCAP–10054–P–A are acceptable for use in determining DCPP Unit 1 and 2 cycle specific core operating limits.

The change does not affect plant operation, or physically alter or change the function of structures, systems, or components required to mitigate the consequences of a design basis accident. In addition, it cannot initiate a transient or affect the probability of occurrence of any previously analyzed accident.

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

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

The proposed change revises the TS to allow the use of NRC approved analytical methods in WCAP–10054–P–A, Addendum 2, Revision 1, and other NRC approved addenda to WCAP–10054–P–A, to determine core operation limits. The change is consistent with the requirements of the TS, and does not affect plant operation, or physically alter or change the function of structures, systems, or components required to mitigate the consequences of a design basis accident.

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

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

The proposed change revises the TS to allow the use of the NRC approved analytical methods in WCAP–10054–P–A, Addendum 2, Revision 1 and other NRC approved addenda to WCAP–10054–P–A, to determine core operating limits. The change is consistent with the requirements of the TS, and does not affect plant operation, or physically alter or change the function of structures, systems, or components required to mitigate the consequences of a design basis accident. The acceptance limits for the small break loss-of-coolant accident are not affected by this change and will continue to be met.

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

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

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

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: January 25, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to relocate the time restriction for movement of irradiated fuel and its related basis page from the TSs to the IP3 Final Safety Analysis Report (FSAR).

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

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

Response

Relocation (i.e., removal from TS) of TS 3.8.A.9 and its basis for the minimum time prior to movement of more than 76 irradiated fuel assemblies (267 hour limit) will not involve a significant increase in the probability or consequences of an accident since the relocation of the TS to administrative controls governed by 10 CFR 50.59 (FSAR) does not affect the availability or function of fuel storage and handling equipment or the SFP [spent fuel pool] cooling system. The waiting time of 267 hours following plant shutdown before unloading more than 76 assemblies from the reactor is to ensure that the maximum SFP water temperature will be within design objectives as stated in the FSAR.

The waiting time of 267 hours is not an initiator of an accident and the proposed change does not alter overall system operation, physical design, system configuration, or operational setpoints. There will be no significant increase in the consequences of an accident because the restricted movement time for irradiated fuel will continue to be administratively controlled under 10 CFR 50.59.

The other TS of section 3.8.A (such as the remaining portion of 3.8.A.9, and 3.8.A. 10) and the other controls ensure that doses from a postulated FHA are within 10 CFR 100 limits.

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

Response

The basis for the waiting time of 267 hours following plant shutdown before unloading more than 76 assemblies from the reactor is to ensure that the maximum pool water temperature will be within design objectives as stated in the FSAR. Relocation of this waiting time of 267 hours for irradiated fuel will not create the possibility of a new or different kind of accident from any previously evaluated. The TS change will not create the possibility of a new or different kind of accident from any previously evaluated since it does not alter the administrative controls for fuel handling or the operation, physical design, system configuration, or operational setpoints for fuel handling and SFP cooling. The plant systems for fuel storage and handling, and SFP cooling are operated in the same manner as before and, consequently, the relocation does not introduce any new accident initiators or failure mechanisms and does not invalidate the existing FHA response. The minimum waiting time for movement of more than 76 irradiated fuel assemblies is not an accident initiator. The minimum waiting time will continue to be controlled under 10 CFR 50.59.

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

Response

Relocation (i.e., removal from TS) of TS 3.8.A.9 and its basis for the waiting time of 267 hours following plant shutdown for irradiated fuel will not involve a significant reduction in margin of safety. The waiting time of 267 hours following plant shutdown before unloading more than 76 assemblies from the reactor is to ensure that the maximum SFP water temperature will be within design objectives as stated in the FSAR. The relocation is a change to the administrative controls that are used to limit the heat load on the SFP cooling system, and those administrative controls will be governed by 10 CFR 50.59. The manner in which fuel storage and handling is performed, and how the SFP cooling system is operated does not change and there is no change to physical design, system configuration, or operational setpoints. The other controls and the existing TS assure that dose from a postulated FHA are within 10 CFR 100 limits. Previous analyses remain unchanged. The current TS does not meet the criteria in 10 CFR 50.36(c)(2)(ii) for inclusion in the Technical Specifications.

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

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

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: January 28, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to change the setpoint of the automatic reactor trip on turbine trip to at or below the P–8 setpoint.

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

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

The addition of reactor trip on turbine trip at [greater than or equal to] 50% to the P–8 Permissive function versus its current setting of [greater than or equal to] 10%, as revised in TS section 2.3.1.C.(3), 2.3.2.A, 2.3.2.B, Table 3.5–2, item 12, Table 4.1–1, item 21 and associated bases, does not significantly increase the probability or consequences of an accident previously evaluated. This additional function, change in reactor trip on turbine trip setpoint, does not cause the initiation of any accident, nor create any new credible limiting single failure, nor result in any event previously deemed incredible being made credible. The existing separation of the reactor and protection functions are not adversely impacted. In addition, the safety functions of safety related systems and component, which are related to accident mitigation, have not been altered. The change in the P–7 or P–8 circuitry does not directly initiate an accident. The consequences of accidents previously [evaluated] in the IP3 FSAR [final safety analysis report] are unaffected by this change because no change to any equipment response or accident mitigation scenario has resulted. There are no additional challenges to fission product barrier integrity. Therefore, the probability or consequences of an accident previously evaluated will not be increased.

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

By adding the reactor trip on turbine trip at [greater than or equal to] 50% to the P–8 Permissive function and setpoint, versus its current setting of [greater than or equal to] 10% and revising TS sections 2.3. l.C.(3), 2.3.2.A, 2.3.2.B, Table 3.5–2, item 12, Table 4.1–1, item 21 and associated bases, does not create the possibility of a new or different kind of accident than any accident already evaluated. The additional function added to the P–8 Permissive does not result in any event previously deemed incredible being made credible. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this change. In addition, the safety functions of safety related systems and components, which are related to accident mitigation, have not been altered. Therefore, the possibility of a new or different kind of accident is not created.

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

The addition of the reactor trip on turbine trip at [greater than or equal to] 50% to the P–8 Permissive function, versus its current setting of [greater than or equal to] 10% and associated changes to TS Sections 2.3. l.C.(3), 2.3.2.A, 2.3.2.B, Table 3.5–2, item 12, Table 4.1–1, item 21 and the associated bases, will have no effect on the availability, operability or performance of the safety-related systems and components and does not affect the plant TS requirements. The current licensing basis safety analyses for IP3 remain bounding with the modification to the P–8 Permissive function; therefore, the margin of safety as defined in the TS is not reduced. The change to the IP3 TS does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: January 28, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 Technical Specifications (TSs) proposes to reduce the number of Emergency Diesel Generators (EDGs) required to be operable during cold shutdown from 2 to 1 under certain conditions.

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

(1) Does the proposed license amendment involve a significant increase in the

probability or consequences of an accident previously [evaluated]?

Response

No. The equipment, which is affected by the proposed Technical Specification change, is not an initiator to those accidents postulated to occur during Cold Shutdown or Refueling operating conditions. A comprehensive systems review and EDG loading electrical analysis has demonstrated the ability of those shutdown support systems, necessary to provide safe shutdown needs, to perform their accident mitigation functions for the postulated accidents during Cold Shutdown and Refueling conditions. One EDG can support the necessary electrical loads required in Cold Shutdown and Refueling in the event of postulated accidents along with a LOOP [loss of offsite power] in the time frame required to prevent reactor core/cavity/SFP [spent fuel pool] heatup concerns. This EDG support relies upon existing plant designed manual closure of 480VAC EDS [electrical distribution system] bus tie breakers to allow a single EDG to pick up other 480VAC EDS bus loads, such as supplying an RHR [residual heat removal] pump and SFP cooling pump, located on 480VAC EDS buses 3A, 5A, or 6A. Together, operability of the required offsite circuit(s) and one EDG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated accidents during shutdown (e.g., Fuel Handling Accidents). Action statements provide prompt, specific guidance to ensure sufficiently conservative plant response should the expected EDG power supply not be available. These Action Statements are similar to those in the STS [Standard Technical Specifications]. Therefore, the proposed license amendment (i.e., changes to 3.7.F.4 and the added sections of 3.7.F.5 & 3.7.F.6) does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

Response

No. The proposed license amendment does not involve any physical changes to plant systems or component set points. The use of 480VAC EDS bus tie breakers to power loads from an energized 480VAC bus is part of present plant design and included within the present LOOP Off-Normal operating procedures when the reactor is in Cold Shutdown operating conditions. As discussed in the Standard Technical Specifications, NUREG 1431, during plant shutdown with one EDG, it is not required to assume a single failure and concurrent loss of all offsite or all onsite power. Worst case bounding events are deemed not credible in Cold Shutdown and Refueling conditions because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and ultimately result in minimal consequences. The lone EDG is capable of accepting and starting required loads within the assumed loading sequence intervals and continue to operate until offsite power can be provided to the 480VAC EDS buses. Action statements provide prompt, specific guidance to ensure sufficiently conservative plant response should the expected EDG power supply not be available. These action statements are similar to those in the STS. Therefore, the proposed license amendment (i.e., changes to 3.7.F.4 and added sections 3.71.5 & 3.7.F.6) does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response

No. The electrical power system specifications support the equipment required to be operable, commensurate with the current level of safety, including the equipment requiring an EDG backed power source. The design review results demonstrate that operation in the conditions of Cold Shutdown and Refueling, in accordance with the proposed Technical Specification change, is acceptable from an accident mitigation standpoint. The basic system functions in Cold Shutdown and Refueling operating conditions are not changed. One EDG can supply the necessary electrical power needs during these plant operating conditions, and in the time frame required to prevent reactor core/cavity/SFP heatup concerns, with sufficient ''kw loading'' to spare. The analysis conducted shows that the systems are capable of performing their design basis functions. Applicable safety analysis in the Standard Technical Specifications, NUREG 1431, discusses these system requirements as well (i.e., it is not required to assume a single failure and concurrent loss of all offsite or all onsite power). Action statements, similar to those in the Standard Technical Specifications, provide prompt, specific guidance to ensure sufficiently conservative plant response should the expected EDG power supply not be available. On this basis, the proposed license amendment (i.e., changes to 3.7.F.4 and added sections 3.7.F.5 & 3.7.F.6) does not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: January 29, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to change the allowable indicated control rod misalignment.

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

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

Response

No. Based on the Westinghouse evaluation in WCAP–14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS Bases is not reduced in any of the IP3 licensing basis accident analysis. Increasing the magnitude of allowed control rod indicated misalignment (in section 3.10.5) is not a contributor to the mechanistic cause of an accident evaluated in the FSAR [Final Safety Analysis Report]. Neither the rod control system nor the rod position indicator function is being altered. Therefore, the probability of an accident previously evaluated has not significantly increased. Because design limitations continue to be met, and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid.

Therefore, the consequences of an accident previously evaluated will not be significantly increased.

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

Response

No. Based on the Westinghouse evaluation in WCAP–14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS Bases is not reduced in any of the IP3 licensing basis accident analysis. Increasing the magnitude of allowed control rod indicated misalignment is not a contributor to the mechanistic cause of any accident. Neither the rod control system nor the rod position indicator function is being altered. Therefore, an accident which is new or different than any previously evaluated will not be created.

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

Response

No. Based on the Westinghouse evaluation in WCAP–14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS Bases is not reduced in any of the IP3 licensing basis accident analysis based on the changes to safety analyses input parameter values as discussed in WCAP–14668. Since the evaluations in Section 3.0 of WCAP–14668 demonstrate that all applicable acceptance criteria continue to be met, the proposed change will not involve a significant reduction in margin of safety.

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

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

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: January 29, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to change the allowable indicated control rod misalignment.

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

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

Response

No. Based on the Westinghouse evaluation in WCAP–14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS Bases is not reduced in any of the IP3 licensing basis accident analysis. Increasing the magnitude of allowed control rod indicated misalignment (in Section 3.10.5) is not a contributor to the mechanistic cause of an accident evaluated in the FSAR [Final Safety Analysis Report]. Neither the rod control system nor the rod position indicator function is being altered. Therefore, the probability of an accident previously evaluated has not significantly increased. Because design limitations continue to be met, and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid.

Therefore, the consequences of an accident previously evaluated will not be significantly increased.

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

Response

No. Based on the Westinghouse evaluation in WCAP–14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS Bases is not reduced in any of the IP3 licensing basis accident analysis. Increasing the magnitude of allowed control rod indicated misalignment is not a contributor to the mechanistic cause of any accident. Neither the rod control system nor the rod position indicator function is being altered. Therefore, an accident which is new or different than any previously evaluated will not be created.

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

Response

No. Based on the Westinghouse evaluation in WCAP–14668, the Authority has determined that all pertinent licensing basis acceptance criteria have been met, and the margin of safety as defined in the TS Bases is not reduced in any of the IP3 licensing basis accident analysis based on the changes to safety analyses input parameter values as discussed in WCAP–14668. Since the evaluations in Section 3.0 of WCAP–14668 demonstrate that all applicable acceptance criteria continue to be met, the proposed change will not involve a significant reduction in margin of safety.

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

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

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: March 22, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.7.1.6, ''Atmospheric Steam Relief Valves,'' and add a new TS for atmospheric steam relief valve instrumentation, to ensure that the automatic feature of the steam generator power-operated relief valve (i.e., atmospheric steam relief valves) remains operable during Modes 1 and 2. In addition, the proposed change would add an associated surveillance requiring that a channel calibration on the steam generator power-operated relief valve be performed every 18 months.

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

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

The methodologies used in the accident analyses remain unchanged. The automatic actuation of the Steam Generator Power Operated Relief Valves is not a new design feature. The effects of the inadvertent opening of a Steam Generator Power Operated Relief Valve are currently analyzed as described in Section 15.1.4 of the Updated Final Safety Analysis Report. The radiological consequences for the Small Break Loss of Coolant Accident (SBLOCA) event presented in the Updated Final Safety Analysis Report remain unchanged. The calculated Peak Clad Temperature is 1849°F remaining substantially below the 2200°F acceptance limit of 10 CFR 50.46. Although the manual control specification is relocated from Specification 3.7.1.6 to the new instrumentation specification, the limiting condition for operation, applicability and action statements for manual controls remain unchanged. Therefore no increase in the probability or consequences of any accident previously evaluated will occur.

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

The automatic actuation of the Steam Generator Power Operated Relief Valves is not an accident initiator for the SBLOCA event. The automatic actuation of the Steam Generator Power Operated Relief Valves currently exists at the South Texas Project and is not a new design feature. The description of the Steam Generator Power Operated Relief Valves currently exists in the Updated Final Safety Analysis Report. This change does not represent a change to the facility and does not affect the safety functions and reliability of systems, structures, or components in any new manner. Operating procedures have a temporary administrative control to ensure the automatic actuation of the Steam Generator Power Operated Relief Valves remains operable in Modes 1 and 2. This condition will become permanent with the approval of this Technical Specification Amendment proposal. Although the manual control specification is relocated from Specification 3.7.1.6 to the new instrumentation specification, the limiting condition for operation, applicability and action statements for manual controls remain unchanged. Since the automatic actuation of the Steam Generator Power Operated Relief Valves is not an accident initiator and is not a new design feature to the facility, no possibility exists for a new or different kind of accident from those previously evaluated.

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

The proposed change results in the calculated Peak Clad Temperature of 1849°F remaining well below the acceptance limit of 10 CFR 50.46 and comparable to the results currently described in the Updated Final Safety Analysis Report. Therefore, the change does not involve a significant reduction in a margin of safety.

Based on the above, the South Texas Project has evaluated the proposed Technical Specification change and determined it does not represent a significant hazards consideration.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of application for amendments: March 19, 1999 (TS 99–01).

Brief description of amendments: The proposed amendments would change the SQN Technical Specifications (TS) for Operating Licenses DPR–77 (Unit 1) and DPR–79 (Unit 2) by relocating TS Sections 3.8.3.1, 3.8.3.2, and 3.8.3.3 to the SQN Technical Requirements Manual. These sections provide requirements for electrical overcurrent isolation devices.

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

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

The proposed revision to the TS relocates the requirements for SQN's electrical equipment protective devices without changing the current requirements. TVA does not consider these devices to be the source of any accident; therefore, this administrative relocation of the requirements will not increase the possibility of an accident. SQN's electrical equipment protective devices will continue to provide fault protection for circuits and equipment. Changes to the relocated requirements will be processed, in accordance with 10 CFR 50.59, to ensure changes are not implemented that would reduce the functionality or introduce an unreviewed safety question to SQN's electrical equipment devices. Therefore, the proposed relocation of the TS requirements for electrical equipment protective devices will not increase the consequences of an accident.

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

SQN's electrical equipment protective devices ensure proper operation of plant equipment. These devices are not associated with accident mitigation or previously evaluated accidents and would not be the initiator of any new or different kind of accident. The proposed change does not alter the current functions of these devices, therefore, this proposed change will not create the possibility of a new or different kind of accident.

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

The requirements for SQN's electrical equipment protective devices are unchanged by the proposed relocation of the requirements to the SQN Technical Requirements Manual. The function of these devices and the surveillance testing to ensure operability of these devices remains unchanged. Any future changes to these requirements will be evaluated, in accordance with 10 CFR 50.59, to ensure acceptability and NRC review as required. Accordingly, the proposed change will not result in a reduction in a 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.

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

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

NRC Section Chief: Sheri R. Peterson.

Notice of Issuance of Amendments to Facility Operating Licenses

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

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

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

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

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: December 16, 1997, as supplemented August 31, and December 7, 1998.

Brief description of amendment: This amendment changes Technical Specification 4.7.1.2.1.a.2.a, Auxiliary Feedwater (AFW) System Surveillance Requirements, by changing the differential pressure and flow requirements of the steam turbine-driven AFW pump to allow testing of the pump at a lower speed.

Date of issuance: April 1, 1999.

Effective date: April 1, 1999.

Amendment No.: 87.

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

Date of initial notice in **Federal Register:** February 11, 1998 (63 FR 6981).

The August 31, and December 7, 1998, submittals contained 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 April 1, 1999.

No significant hazards consideration comments received: No.

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

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: September 1, 1998, as supplemented on March 19, 1999.

Brief description of amendment: This amendment changes Technical Specification (TS) 3/4.9.11, ''Water Level—New and Spent Fuel Pools,'' and its associated Bases by requiring 23 feet of water above the top of fuel rods within irradiated fuel assemblies seated in the storage racks.

Date of issuance: April 8, 1999.

Effective date: April 8, 1999.

Amendment No.: 88.

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

Date of initial notice in **Federal Register:** September 23, 1998 (63 FR 50935).

The March 19, 1999, submittal contained 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 April 8, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 25, 1998.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to support on-line replacement of the Braidwood, Unit 2, vital batteries.

Date of issuance: March 26, 1999.

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

Amendment Nos.: 99 and 99.

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

Date of initial notice in **Federal Register:** February 24, 1999 (64 FR 9185).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 26, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Date of application for amendments: January 21, 1999.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) by relocating TS Section 3/4.6.I, ''Primary System Boundary-Chemistry'' and associated bases to the Updated Final Safety Analysis Report (UFSAR) and to applicable plant procedures.

Date of issuance: March 31, 1999.

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

Amendment Nos.: 187 and 184.

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

Date of initial notice in **Federal Register:** February 24, 1999 (64 FR 9186).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: January 28, 1999.

Brief description of amendments: The amendments revised Technical Specifications Section 3.7.13, ''Fuel Handling Ventilation Exhaust System,'' and associated Bases to correct discrepancies between the current design and this section.

Date of issuance: March 26, 1999.

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

Amendment Nos.: Unit 1–176; Unit 2–168.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 24, 1999 (64 FR 9187).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 26, 1999.

No significant hazards consideration comments received: No.

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

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: October 15, 1998, as supplemented December 15, 1998, and January 11 and 21, 1999.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) to change the heatup, cooldown, and inservice test limitations for the reactor coolant system of each unit to a maximum of 26 effective full-power years. The amendments also revise the TSs for low temperature overpressure protection to reflect the revised pressure-temperature limits of the reactor vessels.

Date of Issuance: March 30, 1999.

Effective date: As of the date of issuance to be implemented within 90 days from the date of issuance.

Amendment Nos.: Unit 1–302; Unit 2–302; Unit 3–302.

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

Date of initial notice in **Federal Register:** December 2, 1998 (63 FR 66592).

The December 15, 1998, and January 11 and 21, 1999, letters provided clarifying information that did not change the scope of the original **Federal Register** notice and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 30, 1999.

No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: October 15, 1998, as supplemented

December 14, 1998, February 18, 1999, and February 23, 1999.

Brief description of amendments: These amendments made several changes that are administrative in nature. The changes (1) made editorial changes that delete obsolete material or material adequately described elsewhere, changed action statement numbers, updated technical specification (TSs) index pages, and made changes to be consistent with the guidance provided in the improved standard technical specifications for Westinghouse reactors (NUREG–1431, Revision 1); (2) deleted reporting requirements that are duplicated in various sections of Title 10 of the Code of Federal Regulations; and (3) relocated the requirement for meteorological monitoring instrumentation from the TSs to the Licensing Requirements Manual.

The February 18, 1999, and February 23, 1999, letters withdrew a portion of the amendment request that would have deleted the description of the site exclusion boundary from the TSs. The description of the site exclusion boundary will remain in the TS.

Date of issuance: March 26, 1999.

Effective date: Units 1 and 2, as of date of issuance, to be implemented within 60 days.

Amendment Nos.: 220 and 97.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications and licenses.

Date of initial notice in **Federal Register:** November 18, 1998 (63 FR 64111).

The December 14, 1998, February 18, 1999, and February 23, 1999, letters did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 26, 1999.

No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of application for amendment: March 10, 1997, as supplemented July 28, 1997, September 17, 1997, April 30, 1998, January 29, 1999, and February 26, 1999.

Brief description of amendment: The amendment modifies Technical Specification 3/4.4.5, ''Steam Generators,'' and its associated Bases and adds a new license condition to Appendix D to allow repair of steam generator tubes by installation of sleeves developed by ABB Combustion Engineering. In addition, the amendment deletes the option for using the kinetic sleeving methodology previously approved for use at Beaver Valley Power Station, Unit 2.

Date of issuance: March 26, 1999.

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

Amendment No: 98.

Facility Operating License No. NPF–73. Amendment revised the Technical Specifications and License.

Date of initial notice in **Federal Register:** April 23, 1997 (62 FR 19829).

The July 28, 1997, September 17, 1997, April 30, 1998, January 29, 1999, and February 26, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the April 23, 1997, **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 1999.

No significant hazards consideration comments received: No.

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

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: January 12, 1999, supersedes application dated May 31, 1996.

Brief description of amendment: The amendment adds an additional required action to the Limiting Condition for Operation (LCO) 3.9.1, ''Refueling Equipment Interlocks,'' of the RBS Technical Specifications. The additional action will allow an alternative to the current action for one or more inoperable refueling equipment interlocks. The current action is to ''suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).'' The alternative action will be to (1) insert a control rod withdrawal block, and (2) verify all control rods are fully inserted in core cells containing one or more fuel assemblies. The amendment also revised the Bases for LCO 3.9.1 actions to describe the alternative action.

Date of issuance: March 26, 1999.

Effective date: March 26, 1999.

Amendment No.: 104.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 10, 1999 (64 FR 6695).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 1999.

No significant hazards consideration comments received: No.

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

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: November 30, 1998.

Brief description of amendment: The amendment changes Technical Specification 3.1.2, ''Liquid Poison System,'' and its associated Bases to correct the required concentration and volume of boron solution.

Date of issuance: April 2, 1999.

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

Amendment No.: 166.

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

Date of initial notice in **Federal Register:** December 30, 1998 (63 FR 71970).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 2, 1999.

No significant hazards consideration comments received: No.

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

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: April 1, 1998, as supplemented May 29, June 26, and August 4, 1998.

Brief description of amendment: The amendment revises the Millstone Unit 3 final safety analysis report (FSAR) by adding a new sump pump subsystem to address groundwater inleakage through the containment basemat.

Date of issuance: March 17, 1999.

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

Amendment No.: 168.

Facility Operating License No. NPF–49: Amendment authorized changes to the FSAR.

Date of initial notice in **Federal Register:** April 22, 1998 (63 FR 19974).

The May 29, June 26, and August 4, 1998, letters provided clarifying information that did not change the scope of the April 1, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment, state consultation, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 17, 1999.

No significant hazards consideration comments received: No public comments received. A petition to intervene was received from the Citizens Regulatory Commission that was dismissed and terminated by the NRC Atomic Safety Licensing Board (LBP–98–22).

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

PECO Energy Company, Public Service Electric and Gas Company Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: February 4, 1998, as revised September 29, 1998.

Brief description of amendments: The amendments revise the Technical Specifications surveillance requirements concerning secondary containment doors.

Date of issuance: April 7, 1999.

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

Amendments Nos.: 227 and 230.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 14, 1998 (63 FR 38202).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 7, 1999.

No significant hazards consideration comments received: No.

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

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: October 22, 1998.

Brief description of amendment: This amendment revises Technical Specification (TS) 4.8.2.1.b.3 to increase the minimum battery electrolyte temperature limit from 60°F to 72°F. This change resolves a discrepancy in the electrolyte temperature assumed in the Class 1E battery sizing calculations versus the limit specified in the TSs.

Date of issuance: March 25, 1999.

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

Amendment No.: 118.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 2, 1998 (63 FR 66602).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 25, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

Date of application for amendments: December 4, 1998.

Brief description of amendments: The amendments make two changes to the TS. The first change revises the Unit 1 TS Section 2.1.1.2 to delete the footnote that specifies that the Safety Limit Minimum Critical Power Ratios are for Cycle 18 only. The second change revises the TS for both units by deleting Section 5.6.5.b.2) and incorporating Section 5.6.5.b.1) into Section 5.6.5.b.

Date of issuance: April 1, 1999.

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

Amendment Nos.: Unit 1–215; Unit 2–156.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 27, 1999 (64 FR 4161).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 1, 1999.

No significant hazards consideration comments received: No.

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

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: October 15, 1998, as supplemented by letter dated November 11, 1998.

Brief description of amendments: The amendments change the Vogtle Electric Generating Plant Unit 1 and 2 Facility Operating Licenses to delete or modify certain license conditions that have become obsolete or inappropriate. In addition, the Technical Specifications and Bases are reissued to reflect new word processing software.

Date of issuance: March 26, 1999.

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

Amendment Nos.: Unit 1–107; Unit 2–85.

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

Date of initial notice in **Federal Register**: December 2, 1998 (63 FR 66602).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 26, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia

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

Date of applications for amendment: October 31, 1997, as supplemented by letter dated September 29, 1998, and application dated July 30, 1998.

Brief description of amendment: The amendment revised Tables 3.3–3, 3.3–4, and 4.3–2 of the technical specifications regarding the engineered safety feature actuation system (ESFAS) Functional Unit 6.f, ''Loss of Offsite Power—Start Turbine-Driven Pump,'' by establishing separate requirements for the analog and digital portions of the associated circuit. The amendment also adds a note to TS Table 4.3–2 to clarify that the verification of time delays associated with ESFAS Functional Units 8.a and 8.b, ''Loss of Power,'' is only performed as part of the channel calibration.

Date of issuance: April 2, 1999.

Effective date: April 2, 1999, to be implemented within 30 days of the date of issuance.

Amendment No.: 130.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 16, 1998 (63 FR 69348).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 2, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Elmer Ellis Library, University of Missouri, Columbia Missouri 65201.

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

Date of application for amendment: November 18, 1998, as supplemented with additional information by letters dated March 1, 1999, and March 9, 1999.

Brief description of amendment: The amendment revises the pressure/temperature limits and the low-temperature overpressure protection requirements in the facility technical specifications.

Date of issuance: April 1, 1999.

Effective date: April 1, 1999.

Amendment No.: 144.

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

Date of initial notice in **Federal Register**: December 30, 1998. (63FR71978)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001.

Dated at Rockville, Maryland, this 14th day of April 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–9839 Filed 4–20–99; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 23785; 812–11218]

American Capital Strategies, Ltd.; Notice of Application

April 14, 1999.

AGENCY: Securities and Exchange Commission (the ''Commission'').

ACTION: Notice of an application for an order under section 61(a)(3)(B) of the Investment Company Act of 1940 (the ''Act'').

SUMMARY OF APPLICATION: Applicant, American Capital Strategies, Ltd., requests an order approving its 1997 Disinterested Director Stock Option Plan (the ''Plan'') and the grant of certain stock options under the Plan.

FILING DATES: The application was filed on July 10, 1998 and amended on November 12, 1998. Applicant has agreed to file an amendment to the application during the notice period, the substance of which is reflected in this notice.

Hearing or Notification of Hearing: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Commission's Secretary and serving applicant with a copy of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on May 10, 1999, and should be accompanied by proof of service on applicant, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a hearing may request notification by writing to the Commission's Secretary.

ADDRESSES: Secretary, Commission, 450 5th Street, NW, Washington, DC 20549–0609. Applicant, c/o Samuel A. Flax, Esquire, Arnold & Porter, 555 Twelfth Street, NW, Washington, DC 20004–1206.

FOR FURTHER INFORMATION CONTACT: Emerson S. Davis, Sr., Senior Counsel, at (202) 942–0714, or George J. Zornada, Branch Chief, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

SUPPLEMENTARY INFORMATION: The following is a summary of the application. The complete application is available for a fee at the Commission's Public Reference Branch, 450 Fifth Street, NW, Washington, DC 20549–0102 (Tel. 202–942–8090).

Applicant's Representations

1. Applicant is a business development company (''BDC'') within the meaning of section 2(a)(48) of the Act.[1] Applicant's primary business is making loans and investments in small and medium-sized companies. Applicant's investment decisions are made by a board of directors (''Board'') based on recommendations of a loan approval committee comprised of senior management. Applicant does not have an external investment adviser within the meaning of section 2(a)(20) of the Act.

2. Applicant requests an order under section 61(a)(3)(B) of the Act approving the Plan, which provides for the grant of options to purchase shares of applicant's common stock to directors who are neither officers nor employees of applicant (''Non-Employee Directors'').[2] Applicant has a nine member Board, the majority of whom are not ''interested persons'' as defined in section 2(a)(19) of the Act. On November 6, 1997, the Board adopted the Plan subject to approval by the Commission and applicant's shareholders. On May 14, 1998, applicant's shareholders approved the Plan. The Plan will not become effective until the date that a Commission order is issued on the application.

3. The Plan provides that each Non-Employee Director will receive an initial grant of options (together with any options issued later under the Plan, ''Options'') to acquire 15,000 shares of applicant's common stock. The Options will vest over a three-year period in 5,000 share increments. Five of the Non-Employee Directors were directors when the Board adopted the Plan. These five Non-Employee Directors will have 5,000 Options vest on November 6 of each of the three years following November 6, 1997. The sixth Non-Employee Director became a director and received an initial grant of 15,000 Options on August 8, 1998. The sixth director's Options will vest in 5,000 increments on August 8th of each of the three following years. Any Options granted prior to the issuance of a Commission order that otherwise would have vested

[1] Section 2(a)(48) defines a BDC to be any closed-end investment company that operates for the purpose of making investments in securities described in sections 55(a)(1) through 55(a)(3) of the Act and makes available significant managerial assistance with respect to the issuers of such securities.

[2] Each Non-Employee Director receives $10,000 per year for each year they serve as a director and $1,000 for each Board or committee meeting attended, plus reimbursement of related expenses.