Scoping Comments

Written comments should be mailed to: Michael T. Lesar, Acting Chief, U.S. Nuclear Regulatory Commission, Rules & Directives Branch, Division of Administrative Services, Office of Administration, Mail Stop T6D59, Washington, DC 20555.

Comments will also be accepted by e-mail. Interested parties may e-mail their comments to teh@nrc.gov. Comments will be accepted by fax at 301–415–5398, Attention: Tim Harris.

NRC will make the scoping summaries and project-related materials available for public review through our electronic reading room: *http://www.nrc.gov/NRC/ADAMS/index.html.* The scoping meeting summaries and project-related materials will also be available on the NRC's MOX web page: *http://www.nrc.gov/NRC/NMSS/MOX/index.html* (case sensitive).

The NEPA Process

The EIS for the MOX Facility will be prepared according to the National Environmental Policy Act of 1969, the Council on Environmental Quality's Regulations for Implementing the Procedural Provisions of NEPA (40 CFR Parts 1500–1508), and NRC's NEPA Regulations (10 CFR Part 51).

The draft EIS is scheduled to be published in February 2002. A 45-day comment period on the draft EIS is planned, and public meetings to receive comments will be held approximately three weeks after distribution of the draft EIS. Availability of the draft EIS, the dates of the public comment period, and information about the public meetings will be announced in the **Federal Register**, on NRC's MOX web page, and in the local news media when the draft EIS is distributed. The final EIS, which will incorporate public comments received on the draft EIS, is expected in September 2002.

Signed in Rockville, MD, this 1st day of March 2001.

For the Nuclear Regulatory Commission.

Charlotte E. Abrams,

Acting Chief, Environmental and Performance Assessment Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 01–5509 Filed 3–6–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of March 5, 12, 19, 26, April 2, 9, 2001.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Weeks of March 5, 2001

There are no meetings scheduled for the Week of March 5, 2001.

Week of March 12, 2001—Tentative

Monday, March 12, 2001

1:25 p.m.

Affirmation Session (Public Meeting) (If needed).

1:30 p.m.

Discussion of Management Issues (Closed-Ex. 2)

Week of March 19, 2001—Tentative

Thursday, March 22, 2001

10:25 a.m.

Affirmation Session (Public Meeting) (If needed).

10:30 a.m.

Meeting with Advisory Committee on Nuclear Waste (ACNW) (Public Meeting) (Contact: John Larkins, 301–415–7360).

This meeting will be webcast live at the Web address—*www.nrc.gov/live.html.*

Week of March 26, 2001—Tentative

There are no meetings scheduled for the Week of March 26, 2001.

Week of April 2, 2001—Tentative

There are no meetings scheduled for the Week of April 2, 2001.

Week of April 9, 2001—Tentative

Monday, April 9, 2001.

1:30 p.m.

Briefing on 10 CFR Part 71 Rulemaking (Public Meeting) (Contacts: Naiem Tanious, 301–415–6103; David Pstrak, 301–415–8486).

Tuesday, April 10, 2001

10:25 a.m.

Affirmation Session (Public Meeting) (If needed).

10:30 a.m.

Meeting on Rulemaking and Guidance Development for Uranium Recovery Industry (Public Meeting) (Contact: Michael Layton, 301–415–6676).

*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: David Louis Gamberoni (301) 415–1651.

Additional Information:

By a vote of 5–0 on February 23, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that "Discussion of Intragovernmental Issues (Closed-Ex. 9)" be held on February 26, 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 the distribution, please contact the Office of the Secretary, Washington, D.C. 20555 (301–415–1969). 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 dkw@nrc.gov.

Dated: March 1, 2001.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 01–5723 Filed 3–5–01; 2:21 pm]

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 February 12, 2001, through February 23, 2001. The last biweekly notice was published on February 21, 2001 (66 FR 11050).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of Administrative 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By April 6, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: December 20, 2000.

Description of amendments request: The amendments would revise the Technical Specifications to incorporate changes required to support operation with replacement steam generators. The proposed changes will (1) accommodate geometric differences between the original and replacement steam generators, (2) increase the reactor coolant flow rate from the current value which was recently established to accommodate more tube plugging, and (3) delete tube sleeving options approved for the original steam generators.

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

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

A. Technical Specification Table 3.3.1–1, Item 7

Technical Specification Table 3.3.1–1, ''Reactor Protective System Instrumentation,'' Item 7 sets the allowable value for ''Steam Generator Level-Low'' function to greater than or equal to 10 inches below the top of the feed ring. To accommodate the geometric difference in the location of the top of the feed ring with respect to the pedestal between the original steam generators (OSG) (510.8 inches) and the replacement steam generators (RSG) (484.8 inches), the proposed amendment would change the allowable value for ''Steam Generator Level-Low'' function to greater than or equal to 50 inches below normal water level. Since normal water levels for RSG and OSG with respect to the pedestal are identical and the current steam generator level-low reactor trip setpoint ''≥10 inches below top of feed ring'' is ''≥ 50 inches below normal water level'' for both the RSG and OSG, the functionality of the steam generator level-low reactor trip setpoint will be unchanged. Furthermore, use of normal water level as the point of reference instead of top of the feed ring is more practical and appropriate since it is the frame of reference for steam generator water level indication used in the Control Room by the operators.

The design basis accident affected by the proposed change is the Loss of Feedwater Flow event. The Steam Generator Level-Low Reactor Trip Setpoint, in combination with the Auxiliary Feedwater Actuation System, ensures that adequate secondary side water inventory exists in both RSGs to remove decay heat following a Loss of Feedwater Flow event. To ensure that the acceptance criteria for the Loss of Feedwater Flow event are met with the RSGs, there must be at least as much mass in RSG at the Safety Analysis water level as in the OSG. The OSG Safety Analysis water level is 116.4 inches below normal water level. Using the same method to predict steam generator inventory, at this water level, OSG has 64,049 Ibm water mass and RSG has 64,115 Ibm water mass. Therefore, the RSG has more post-reactor trip secondary side inventory than the OSG which ensures the Loss of Feedwater event acceptance criteria are not challenged.

Therefore, the proposed revision to change the reference setpoint for steam generator low level reactor trip function will not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. LCO [limiting condition for operation] 3.4.1 and Surveillance Requirement 3.4.1.3

The proposed amendment would revise Technical Specification LCO 3.4.1 and Surveillance Requirement 3.4.1.3 to increase reactor coolant minimum required total flow rate back to the originally established value of 370,000 gpm [gallons per minute] from the current value of 340,000 gpm, which was recently established to accommodate more tube plugging in the OSG. The flow resistance of the RSG is equivalent to that of the OSG with zero plugged tubes. Therefore, the required minimum RCS [reactor coolant system] total flow rate can be increased to the value previously established for the original steam generators with zero plugged tubes, 370,000 gpm.

Increasing the required minimum RCS total flow rate has no adverse impact on the safety analysis. Crediting more RCS flow in the safety analysis allows for greater flexibility in core design and operation. The increase in RCS flow associated with the RSG is within the bounds previously analyzed for the OSG. The hydraulic forces experienced around the RCS loop, including the core uplift force, are acceptable. The change is more restrictive in nature in that more RCS flow will be required to meet Surveillance Requirement 3.4.1.3 and more RCS flow ensures enhanced core heat removal. The overall core thermal margin in the safety analysis will remain essentially the same.

Therefore, the proposed revision to increase reactor coolant minimum required total flow rate will not involve a significant increase in the probability or consequences of an accident previously evaluated.

C. Technical Specification Administrative Control 5.5.9

The proposed revision deletes three sleeving options from Administrative Technical Specification 5.5.9. The sleeving options are: Westinghouse Laser Welded sleeves, Asea Brown Boveri, Inc. (ABB)-Combustion Engineering Leak Tight sleeves, and the ABB-Combustion Engineering Alloy 800 Leak Limiting sleeves. One of the differences between the OSG and the RSG design is the use of thermally-treated Alloy 690 tube material instead of high temperature mill-annealed Alloy 600 used for the OSG. The three sleeving tube repair options described in Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specification

Administrative Control 5.5.9, are designed specifically for the OSGs' mill-annealed Alloy 600 tubes.

The three sleeving options were acquired by CCNPP for economic reasons to maintain OSG thermal output by minimizing the number of tubes plugged. Therefore, deletion of these repair options from Administrative Control 5.5.9 has no safety significance.

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

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

A. Technical Specification Table 3.3.1–1, Item 7

The RSGs are equivalent in function to the OSGs. Changing Technical Specification Table 3.3.1–1, Item 7 is required to provide a correct and practical reference point from which to measure the Reactor Trip Steam Generator Level-Low Setpoint. As described above in Item 1, the normal water levels for RSG and OSG with respect to the pedestal are identical and the current steam generator level-low reactor trip setpoint, ''≥ 10 inches below top of feed ring'' is ''≥ 50 inches below normal water level'' for both the RSG and OSG. Hence, the functionality of the reactor trip steam generator level-low setpoint will be unchanged. Furthermore, use of normal water level as the point of reference instead of top of the feed ring is more practical and appropriate since it is the frame of reference for steam generator water level indication used in the Control Room by the operators.

Therefore, the proposed revision to change the reference setpoint for steam generator low level reactor trip function will not create the possibility of a new or different [kind] of accident from any accident previously evaluated.

B. LCO 3.4.1 and Surveillance Requirement 3.4.1.3

As described above in Item 1, increasing the required minimum RCS total flow rate has no adverse impact on the plant's safety analyses. The increase in RCS flow associated with the RSG is within the bounds previously analyzed for the OSG. The hydraulic forces experienced around the RCS loop, including the core uplift force, are acceptable. The change is more restrictive in nature in that more RCS flow will be required to meet Surveillance Requirement 3.4.1.3 and more RCS flow ensures enhanced core heat removal.

Therefore, the proposed revision to increase reactor coolant minimum required total flow rate will not create the possibility of a new or different type of accident from any accident previously evaluated.

C. Technical Specification Administrative Control 5.5.9

As described in Item I above, the three sleeving options were acquired by CCNPP for economic reasons to maintain OSO thermal output by minimizing the number of tubes plugged. Therefore, deletion of these repair options from Technical Specification Administrative Control 5.5.9 has no safety significance.

Therefore, the proposed revision will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in the margin of safety

A. Technical Specification Table 3.3.1–1, Item 7

As described above in Item 1, the design basis accident affected by the proposed change is the Loss of Feedwater Flow event. The Steam Generator Level-Low Reactor Trip Setpoint, in combination with the Auxiliary Feedwater Actuation System, ensures that adequate secondary side water inventory exists in both RSGs to remove decay heat following a Loss of Feedwater Flow event. To ensure that the acceptance criteria for the Loss of Feedwater Flow event are met with the RSGs, there must be at least as much mass in RSG at the Safety Analysis water level as in the OSG. The OSG Safety Analysis water level is 116.4 inches below normal water level. Using the same method to predict steam generator inventory, at this water level, OSO has 64,049 lbm water mass and RSG has 64,115 Ibm water mass. Therefore, the RSG has more post-reactor trip secondary side inventory than the OSG which ensures the Loss of Feedwater event acceptance criteria are not challenged.

Therefore, the proposed revision to change the reference setpoint for steam generator low level reactor trip function does not involve a significant reduction in the margin of safety.

B. LCO 3.4.1 and Surveillance Requirement 3.4.1.3

As described above in Item 1, increasing the required minimum RCS total flow rate has no adverse impact on the safety analysis. Crediting more RCS flow in the safety analysis allows for greater flexibility in core design and operation. The increase in RCS flow associated with the RSG is within the bounds previously analyzed for the OSG. The hydraulic forces experienced around the RCS loop, including the core uplift force, are acceptable. The change is more restrictive in nature in that more RCS flow will be required to meet Surveillance Requirement 3.4.1.3 and more RCS flow ensures enhanced core heat removal. The overall core thermal margin in the safety analysis will remain essentially the same.

Therefore, the proposed revision to increase reactor coolant minimum required total flow rate does not involve a significant reduction in the margin of safety.

C. Technical Specification Administrative Control 5.5.9

As described in Item 1C above, the three sleeving options were acquired by CCNPP for economic reasons to maintain OSG thermal output by minimizing the number of tubes plugged. Therefore, deletion of these repair options from Technical Specification Administrative Control 5.5.9 has no safety significance.

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

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: January 26, 2001.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Surveillance Requirement (SR) 3.7.9.2, ''Ultimate Heat Sink (UHS),'' by increasing the maximum allowable temperature of Lake Michigan water from 81.5 °F to 85 °F. The licensee also proposes to reflect this change in the associated TS Bases.

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

The following evaluation supports the finding that operation of the facility in accordance with the proposed changes would not:

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

The UHS is Lake Michigan which is completely passive and is not an accident initiator in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The UHS, by design, mitigates the consequences of accidents by supplying a repository for the decay heat and other excess energy removed in the process of cooling the plant equipment. The safety analysis has been revised to use a maximum UHS water temperature of 85 °F. The results of these revised analyses still meet all of the required acceptance criteria. Therefore, the proposed changes do not affect any of the results of the FSAR [Final Safety Analysis Report] Chapter 14 accident analyses. Hence the consequences of accidents previously evaluated do not change.

Therefore, operation of the facility in accordance with the proposed changes to the Technical Specifications would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change would not alter the design, configuration, or method of operation of the plant. The proposed temperature limit has been verified to be acceptable for UHS operability determinations by its documented use in plant equipment design considerations, and in the FSAR Chapter 14 accident analyses. Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not create the possibility of a new or different kind of accident from any previously evaluated.

c. Involve a significant reduction in the margin of safety.

The proposed change to the Technical Specifications would impose temperature limits already in use in equipment designs and as an initial assumption of the plant accident analyses. The proposed SR limit has been utilized in the accident analyses since 1994. The results of these accident analyses meet all of the required acceptance criteria when using the 85 °F UHS water temperature limit. Therefore, the proposed change to the Technical Specifications would 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.

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: February 12, 2001.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 5.6.5b, ''Reporting Requirements—Core Operating Limits Report (COLR),'' by adding a reference to the existing references of approved analytical methods for determining 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:

The following evaluation supports the finding that operation of the facility in accordance with the proposed changes would not:

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

The proposed change to the list of methodology documents in Specification 5.6.5.b. would not increase the probability or consequence of an accident previously evaluated. Accidents previously evaluated will be unaffected by the addition of a methodology reference because they were analyzed using approved methods. The results of these event analyses met their respective acceptance criteria.

Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the list of methodology documents in Specification 5.6.5.b. would not create the possibility of a new or different accident than previously analyzed. The proposed change only adds an approved methodology document. All accidents remain analyzed using applicable NRC approved methodologies.

Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Involve a significant reduction in the margin of safety.

The proposed change to the list of methodology documents in Specification 5.6.5.b. would not reduce the margin of safety. Because all analyses use approved methodologies and their results satisfy their respective acceptance criteria, the margin of safety is not reduced.

Therefore, the proposed change to the Technical Specifications would 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.

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201

NRC Section Chief: Claudia M. Craig.

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 24, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to incorporate the provisions to perform routine diesel generator (DG) monthly testing by gradually accelerating the DG to operating speed, as opposed to requiring the DG to attain rated voltage and frequency within 10 seconds for DG 1A and DG 1B, and within 13 seconds for DG 1C. In addition, a new TS would be added to require fast start tests of the DGs on a 184-day frequency.

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

The proposed changes affect the surveillance requirements for the emergency diesel generators. The emergency diesel generators are onsite standby power sources intended to provide redundant and reliable power to ESF [engineered safety feature] systems credited as accident mitigating features in design basis analyses. As discussed in Regulatory Guide (RG) 1.9, Revision 3, the proposed changes are intended to allow slower starts of the diesel generators during testing in order to reduce diesel generator aging effects due to excessive testing conditions. As such, the proposed changes should result in improved diesel generator reliability and availability, thereby providing additional assurance that the diesel generators will be capable of performing their safety function. The method of starting the emergency diesel generators for testing purposes does not affect the probability of any previously evaluated accident. Although the changes allow slower starts for the monthly tests, the more rapid start function assumed in the accident analysis is unchanged and will be verified on a 184 day frequency. Therefore the accident analysis consequences are not affected.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes affect the surveillance requirements for the onsite ac [alternating current] sources, i.e. the diesel generators. Accordingly, the proposed changes do not involve any change to the configuration or method of operation of any plant equipment that could cause an accident. In addition, no new failure modes have been created nor has any new limiting failure been introduced as a result of the proposed surveillance changes.

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

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

The proposed changes are intended to bring the existing RBS [River Bend Station] TS requirements for the onsite ac sources in line with regulatory guidance. Under the proposed changes, the emergency diesel generators will remain capable of performing their safety function, and the effects of aging on the diesel generators will be reduced by eliminating unnecessary testing. The diesel generator start times assumed in the current accident analyses are unchanged and will be verified on a 6-month frequency.

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

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: February 5, 2001.

Description of amendment request: The proposed amendment would change the Safety Limit Minimum Critical Power Ratio (SLMCPR) in Technical Specification (TS) 2.1.2 from 1.08 to 1.06. The proposed amendment would also change the parenthetical statements after certain references listed in TS 5.6.5.b to clarify that the analytical methods described in General Electric Nuclear Energy documents inclusive of the latest amendment or revision are used to determine core operating limits. Also, the proposed amendment would add a new reference to TS 5.6.5.b.

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

1. The proposed changes to technical specification do not involve a significant increase in the probability of an accident previously evaluated.

The proposed Safety Limit MCPR (SLMCPR), and its use to determine the Cycle 14 thermal limits, have been derived using NRC approved methods [See application dated February 5, 2001]. These methods do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

The basis of the SLMCPR is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR preserves the margin to transition boiling, and the probability of fuel damage is not increased.

Therefore, the proposed changes to technical specifications do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes result only from revised methods of analysis for the Cycle 14 core reload. These methods have been reviewed and approved by the NRC, do not involve any new or unapproved method for operating the facility, and do not involve any facility modifications. No new initiating events or transients result from these changes.

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

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

The margin of safety will remain the same. The new SLMCPR was derived using NRC approved methods which are in accordance with the current fuel design and licensing criteria. The SLMCPR remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, which is the current margin of safety used to preserve the fuel cladding integrity.

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

Based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599

NRC Section Chief: James W. Clifford.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: February 16, 2001.

Description of amendment request: This amendment would substitute a surveillance interval of ''Once/

Operating Cycle'' for the current surveillance interval of ''Each Refueling Outage,'' for the following instruments in Technical Specification Table 4.2.F: Containment High Radiation Monitor, Reactor Building Vent Radiation Monitor, Main Stack Vent Radiation Monitor, and Turbine Building Vent Radiation Monitor.

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

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

There are no physical changes to Pilgrim being introduced by the proposed changes to the specified instruments. The proposed changes do not modify Pilgrim, i.e., there are no changes in operating pressure, materials or seismic loading. No plant safety limits, setpoints, or design parameters are adversely affected by the proposed changes. The proposed changes do not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected. The proposed changes do enlarge the opportunity-period for performing the subject calibrations by substituting one established Technical Specification definition for another; hence, the proposed changes are administrative in nature because they do not change any methodology, interval, configuration or equipment at Pilgrim.

Thus, the proposed changes do not affect any significant parameter associated with the instruments or calibration interval; therefore, the ability of the instruments to perform their designed safety function is maintained. The change does not impact plant operation. Consequently, operating Pilgrim in conformance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change substitutes one Technical Specification definition for another concerning certain radiation-monitoring instruments. The ability of these instruments to perform their designed-function is not affected by this change, and the surveillance interval remains nominally 24 months. No new modes of operation are introduced by the proposed changes. No plant safety limits, setpoints, or design parameters are herein proposed, nor is any adverse consequence introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes entail the substitution of one Technical Specification definition for another concerning radiation-monitoring instruments. This is an administrative change because such substitution does not modify the operation, configuration, or processes of Pilgrim, nor does the change modify the nominal 24-month surveillance/calibration interval currently in force for these instruments.

The substitution of one Technical Specification definition for another concerning radiation monitoring instruments potentially reduces personnel exposure from calibration-source radiation because site population is less during non-refueling periods. No plant safety limits, setpoints, or design parameters are changed, nor is any adverse consequence introduced by the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599

NRC Section Chief: James W. Clifford.

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

Date of amendment request: February 6, 2001

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) associated with the reactor coolant system (RCS) leakage detection systems, to make them consistent with the requirements in NUREG–1432, ''Standard Technical Specifications, Combustion Engineering Plants.''

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

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

The aforementioned revisions do not involve any physical change to plant design. Relocating the requirements associated with the RCS Leak Detection System from various TSs to ANO–2 [Arkansas Nuclear One, Unit 2] Specification 3.4.6.1 is administrative in nature and does not affect the accident analyses. The RCS water inventory balance is more accurate than normal leak detection methods in regard to actual RCS leak rates, and therefore is an excellent alternative when other leak detection components may become inoperable. Since the proposed changes only affect the requirements for the detection of RCS leakage, the probability that an accident previously evaluated will occur remains unchanged. The proposed changes do not prevent nor limit the diversity of acceptable detection of RCS leakage and, therefore, do not significantly affect the consequences of an accident previously evaluated since leak rate information will remain available to station personnel. Although the non-administrative revisions result in less restrictive requirements, the proposed changes remain within the acceptability of General Design Criteria (GDC) 30 of Appendix A to 10 CFR [Part] 50 and Regulatory Guide (RG) 1.45, and are consistent with the philosophies of the RSTS [Revised Standard Technical Specifications].

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

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The aforementioned revisions do not involve any physical change to plant design. Relocating the requirements associated with the RCS Leak Detection System from various TSs to ANO–2 Specification 3.4.6.1 is administrative in nature and does not affect the accident analyses. The RCS water inventory balance is more accurate than normal leak detection methods in regard to actual RCS leak rates, and therefore is an excellent alternative when other leak detection components may become inoperable. The proposed changes do not prevent acceptable detection of RCS leakage by diverse methods. The detection of a RCS leak does not cause an accident or prevent an accident from occurring. Likewise, detecting a RCS leak while in its initial stages does not create the possibility of a new or different kind of accident than any previously analyzed. Therefore, a new or different kind of accident than that previously analyzed is not expected to result due to the proposed changes of this submittal. Although the non-administrative revisions result in less restrictive requirements, the proposed changes remain within the acceptability of General Design Criteria (GDC) 30 of Appendix A to 10 CFR [Part] 50, Regulatory Guide (RG) 1.45, and are consistent with the philosophies of the RSTS.

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

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

The aforementioned revisions do not involve any physical change to plant design. Relocating the requirements associated with the RCS Leak Detection System from various TSs to the ANO–2 Specification 3.4.6.1 is administrative in nature and does not affect the margin of safety. The RCS water inventory balance is more accurate than normal leak detection methods in regard to actual RCS leak rates, and therefore is an excellent alternative when other leak detection components may become inoperable. Maintaining diverse and accurate RCS leak detection methods available helps to ensure RCS leaks will be detected within an acceptable period of time and, therefore, the proposed changes do not significantly reduce the margin to safety. Although the non-administrative revisions result in less restrictive requirements, the proposed changes remain within the acceptability of General Design Criteria (GDC) 30 of Appendix A to 10 CFR [Part] 50 and Regulatory Guide (RG) 1.45, and are consistent with the philosophies of the RSTS.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: January 17, 2001

Description of amendment request: The licensee proposes to revise the Technical Specifications (TS) requirements for the Emergency Diesel Generator (EDG) 24-hour surveillance test run. Currently, the TS restrict performance of this test to shutdown periods due to historical concerns regarding the effects of a potential failure while the EDGs are paralleled to the off-site power system. The proposed amendment would allow the surveillance test to be conducted with the plant on-line. The licensee has performed an analysis, which shows that conducting the 24-hour EDG test run with the plant on-line results in a very small change in core damage frequency, and is acceptable under the guidelines of Regulatory Guide 1.174. The risks incurred by performing the test on-line will be substantially offset by plant benefits associated with avoiding unnecessary plant transitions and/or reducing risks during shutdown operations.

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

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated for the following reasons:

The change relocating the ''during shutdown'' requirement from TS 4.8.1.1.2.e to the individual surveillance requirements under TS 4.8.1.1.2.e is strictly administrative in nature. Therefore, it does not involve any increase in the probability or consequences of an accident previously evaluated.

For the change that revises Unit 1 TS 4.8.1.1.2.e.6 to remove the restriction to perform the EDG 24-hour endurance test during shutdown, the emergency diesel generators (EDG) and their associated emergency busses are not accident initiating equipment. Therefore, there will be no impact on any accident probabilities by the approval of this amendment. The design of this equipment is not being modified by these proposed changes. In addition, the ability of the EDGs to respond to a design basis accident will not be significantly impacted by these proposed changes. Consequences are no different than presently when an EDG is out-of-service in the current TS allowed outage time during operation in Modes 1 and 2.

Therefore, performing the EDG 24-hour endurance test in Modes 1 and 2 does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated for the following reasons:

No new accident causal mechanisms are created as a result of this amendment request. Equipment will be operated in the same configuration with the exception of the plant Mode in which testing is conducted. No changes are being made to the plant which introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact accident mitigating systems.

The changes removing the restriction to perform the tests during shutdown for Unit 1 TS 4.8.1.1.2.e.6, in its simplest form, is just a request to extend the amount of time the EDG is synchronized to the grid in Modes 1 and 2 from approximately 18 hours (one hour per month) to approximately 42 hours per cycle. The existing surveillance requirement TS 4.8.1.1.2.a.5 requires, in part, that every 31 days each EDG be demonstrated operable by synchronizing to the grid for at least an hour. It is simply a time extension of the existing surveillance requirement. Therefore, performing the EDG 24-hour endurance test in Modes 1 and 2 does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Use of the modified specification would not involve a significant reduction in a margin safety.

The AC electrical distribution system has been designed to provide sufficient redundancy and reliability to ensure the availability of the EDGs to provide the required safety function under design basis events to protect the power plant, the public, and plant personnel.

The proposed changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analysis to establish any margin of safety. PSA evaluations were used to evaluate these changes, and these evaluations determined that the changes are not risk significant. The proposed activity involves changes to the allowed plant mode for the performance specific Technical Specification surveillance requirements.

During the performance of the EDG endurance surveillance test for a 24-hour period, at least one EDG will be available and will adequately respond within the time necessary to mitigate anticipated operational occurrences or postulated design basis accidents.

The calculated total change in CDF, including the conservatively estimated fire risk contribution, is less than 1E–06 per reactor year and the calculated total change in the LERF, including the conservatively estimated fire risk contribution, is less than 1E–07 per reactor year. The change in CDF and LERF is, therefore, within Region III of Regulatory Guide 1.174 Figures 3 and 4, and is considered very small. When the full scope of plant risk is considered, the risks incurred by performing the EDG 24-hour surveillance test during power operation will be substantially offset by plant benefits associated with avoiding unnecessary plant transitions and/or reducing risks during shutdown operations.

The proposed change does not involve a change to the plant design or operation, and thus, does not affect the design of the EDGs, the operational characteristics of the EDGs, the interfaces between the EDGs and other plant systems, or the function or reliability of the EDGs. Because EDG performance and reliability will continue to be ensured by the proposed Technical Specification changes, the proposed changes do not result in a significant reduction of the margin of safety.

Based on the above, FPL has determined that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety; and therefore, does not involve a significant hazards consideration.

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

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

NRC Section Chief: Richard P. Correia.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: January 4, 2001.

Description of amendment request: The proposed amendment requests NRC's approval of the Maine Yankee Atomic Power Company's (MYAPC) Security Plan, Training and Qualification Plan, and Contingency Plan. These plans reflect the addition of provisions related to the loading and storage of spent fuel into the independent spent fuel storage installation (ISFSI) under construction on owner-controlled property adjacent to the plant site.

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 approved Security Plan, or Defueled Security Program, currently implemented is not being changed. The FIT [Fuel in Transit] Security Program and the ISFSI Security Program are being added to the scope of the overall security scheme at the Maine Yankee site. The additions to the overall plan have been evaluated in accordance with 10 CFR 50.54(p) and 10 CFR 72.212(b)(4) and it has been determined that the implementation of the ISFSI and FIT Security Programs would not decrease the effectiveness of the Defueled Security Program, the Defueled Security Guard Training and Qualification Program, or the first four categories of the Defueled Safeguards Contingency Program.

The Defueled Security Program Staffing will be augmented as and if necessary to support Fuel in Transit evolutions. The ISFSI Security Program staffing will be separate from and parallel to the staffing requirements of the Defueled Security Program.

The operational and physical venues of the Defueled Security Program, the FIT Security Program, and the ISFSI Security Program are separate and distinct. The line of demarcation between the three programs is clearly defined and not overlapping. The implementation of any of the programs therefore does not degrade or inhibit the implementation of the other two programs.

The Defueled Program Guard Training and Qualification Plan and the Defueled Safeguards Contingency plan also have not been changed. A separate and parallel ISFSI Training and Qualification Plan and Contingency Plan is included in the ISFSI Security Program. The FIT program uses the Defueled Program, Training and Qualification Plan and Contingency Plan. The physical protection systems described in the ISFSI and FIT Programs are designed to protect against the loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose as described in 10 CFR 72.106.

Therefore, the ISFSI Program revisions of the Security Plan, Guard Training and Qualification Plan and the Safeguards Contingency Plan will not increase the probability or the consequences of an accident previously evaluated since the previously approved Defueled Training and Qualification Plan and Contingency Plan remain unchanged.

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

The FIT and ISFSI Security Programs have no impact on the existing Defueled Security Program since they operate in different physical and licensing venues. The accidents considered for the Spent Fuel Pool, the venue of the Defueled Security Program, are described in the Maine Yankee Defueled Safety Analysis Report. The accidents considered for the FIT and ISFSI are contained in the NAC International, Inc. Final Safety Analysis Report for the UMS Universal Storage System Docket No. 72–1015.

The FIT and ISFSI Security Programs have been crafted to meet or exceed all of the assumptions of the NAC International FSAR concerning accident analyses and the programs meet or exceed all of the applicable requirements of 10 CFR 73.55 with approved exceptions or approved alternative measures. The physical protection systems described in the ISFSI and FIT Programs are designed to protect against the loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose as described in 10 CFR 72.106.

The proposed action does not affect plant systems, structures or components within the venue of the existing Security Plan. The ISFSI and FIT program additions to the Security Plan, Guard Training and Qualification Plan and the Safeguards Contingency Plan do not create the possibility of a new or different kind of accident from any accident previously evaluated since the previously approved Defueled Security Plan, Training and Qualification Plan and Contingency plan remain as is, unaltered.

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

The addition of a separate, parallel ISFSI and FIT Safeguards Program, Training and Qualification Plan, and Contingency Plan does not alter or reduce the effectiveness of the previously approved Defueled Program. The physical protection systems described in the ISFSI and FIT Programs are designed to protect against the loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose as described in 10 CFR 72.106. Therefore, the margin of safety will not be reduced as a result of the ISFSI and FIT additions to the Security Plan, or an ISFSI specific addition of a Guard Training and Qualification Plan or an ISFSI specific addition of a Safeguards Contingency Plan

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

Attorney for licensee: Joseph Fay, Esquire, Maine Yankee Atomic Power Company, 321 Old Ferry Road, Wiscasset, Maine 04578.

NRC Section Chief: Michael T. Masnik.

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

Date of amendment requests: April 17, 2000, as supplemented February 2, 2001.

Description of amendment requests: The proposed amendments would change the Technical Specifications (TSs) for the removal of boric acid storage tanks (BASTs) from the safety injection (SI) system. These changes would accomplish two objectives: (1) Eliminate high concentration boric acid from the SI system and (2) align this specific Prairie Island TS section with the Standard TSs.

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

(1) The proposed amendment will not involve a significant increase in the probability or consequences of accidents previously evaluated.

The proposed change to the CVCS [chemical volume control system] and SI system (increasing the concentration of boric acid in the RWST [refueling water storage tank] and eliminating the BAST as a suction source, respectively) and elimination of or change to associated Technical Specifications do not affect accident initiation. None of the equipment being removed from Sections 3.2 or 3.5 of Technical Specifications are accident initiators. Thus, the proposed changes will not significantly increase the probability of an accident previously evaluated.

Consequences are evaluated in terms of off-site and on-site (control room personnel) dose. Loss of coolant accident (LOCA) dose is unaffected by the proposed changes because the LOCA analysis input assumptions are not changed by the changes proposed in this amendment request. The approved steam line break (SLB) methodology (approved by the NRC in letter dated January 19, 2000) and the expected dose are unaffected by the proposed change.

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

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

The proposed changes to the plant and its Technical Specifications do not introduce any new accident initiators. The proposed changes reduce the number of automatic component actuations needed to support Safety Injection accident mitigation functions. The proposed changes also remove the Technical Specification requirements for the balance of the CVCS components. These requirements were in Technical Specifications to support the boration function of CVCS; however, all boration functions can be met by the safety-related SI system. All the other functions of the CVCS are either backed up by a safety related system or are not required to preclude an accident (reference NSP [Northern States Power] letter of June 14, 1995 and NRC letter of January 8, 1996).

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

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

The proposed changes do not significantly impact the plant response to an accident with respect to the ability to protect fission product barriers. The proposed changes will not result in any significant increase in fuel cladding damage in the event of a postulated accident (accident analyses show the proposed changes meet all acceptance criteria related to maintaining cladding integrity). The proposed changes will not reduce the integrity of the RCS [reactor coolant system] (reduction of boric acid concentrations in the SI systems will not promote any degradation of the components that make up the RCS pressure boundary). The proposed changes will not result in a reduction in containment integrity in the event of a postulated accident (the changes proposed by this amendment do not change the results of the accident analyses with respect to containment response.)

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

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

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

NRC Section Chief: Claudia M. Craig.

Sacramento Municipal Utility District (SMUD), Docket No. 50–312, Rancho Seco Nuclear Station, Sacramento County, California

Date of amendment request: October 23, 2000.

Description of amendment request: The proposed amendment (PA–194) as supplemented by SMUD letter to the USNRC dated January 11, 2001, would change the Permanently Defueled Technical Specification (PDTS) by deleting the definitions for ''site boundary'' and ''unrestricted area;'' revising the definition of the ''site;'' deleting figures D5.1–1, ''Emergency Planning Zone,'' D5.1–2, ''Site Boundary for Gaseous Effluent,'' and D5.1–3, ''Site Boundary for Liquid Effluent;'' and making editorial changes to the other PDTSs because of the above proposed changes. The information proposed for removal from the PDTS is contained in or will be relocated to other licensee-controlled documents.

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:

SMUD has reviewed the proposed PDTS change against each of the criteria in 10 CFR 50.92 and has concluded that the amendment request involves no significant hazards consideration. The following provides SMUD's analysis of the issue of no significant hazards consideration:

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

No. The proposed changes are administrative and involve deleting the definitions of SITE BOUNDARY and UNRESTRICTED AREA from the DEFINITIONS section, revising the definition of the site in Section 5.1 ''SITE,'' deleting all three figures from the DESIGN FEATURES section [SMUD proposes, as described in its January 11, 2001, letter, that these or equivalent figures will be relocated to either the Emergency Plan or the Offsite Dose Calculation Manual, as appropriate], revising Sections D6.8.3.a(2) and D6.8.3.a(4) so that the term ''unrestricted area'' is lower case, and revising Sections D6.8.3.a(8), D6.8.3.a(9), D6.8.3.a(10), and D6.8.3.b(2) so that the term ''site boundary'' is lower case.

These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. Safety limits, limiting safety system settings, and limiting control systems are no longer applicable to Rancho Seco Technical Specifications in the permanently defueled mode, and are therefore not relevant.

The proposed changes do not affect the emergency planning zone, the boundaries used to evaluate compliance with liquid or gaseous effluent limits, and have no impact on plant operations. Therefore, the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. As described above, the proposed changes are administrative. The safety analysis for the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid.

The operating procedures and emergency procedures are not affected. The proposed changes do not affect the emergency planning zone, the boundaries used to evaluate compliance with liquid or gaseous effluent limits, and have no impact on plant operations. Consequently, no new failure modes are introduced as the result of the proposed changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. As described above, the proposed changes are administrative. There are no changes to the design or operation of the facility. The proposed changes do not affect the emergency planning zone, the boundaries used to evaluate compliance with liquid or gaseous effluent release limits, and have no impact on plant operations. Accordingly, neither the design basis nor the accident assumptions in the Defueled Safety Analysis Report (DSAR), nor the Technical Specification Bases are affected. 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 requested amendment involves no significant hazards consideration.

Attorney for licensee: Dana Appling, Esq., Sacramento Municipal Utility District, P.O. Box 15830, Sacramento, California 95852–1830.

NRC Section Chief: Michael T. Masnik.

Southern Nuclear Operating Company, Inc, Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: August 25, 2000.

Description of amendment request: The proposed amendments would revise the Updated Final Safety Analysis Report (UFSAR) described offsite dose analyses based on changes to the letdown flow rate and iodine spike postulated concurrent with a Main Steam Line Break or a Steam Generator Tube Rupture.

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 do not significantly increase the probability or consequences of an accident previously evaluated in the UFSAR. The comprehensive engineering review included evaluations or re-analysis of all accident analyses. Calculations for letdown flow measurement and indication have verified the acceptability of the analyzed letdown flow rate. The letdown flow rate does not initiate any accident; therefore, the probability of an accident has not been increased. All dose consequences have been analyzed or evaluated with respect to the proposed changes, and all acceptance criteria continue to be met. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the UFSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. Therefore, all accident analyses criteria continue to be met and these changes do 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 proposed changes do not involve a significant reduction in a margin of safety. All analyses and evaluations using letdown flow rate as an input have been revised to reflect the proposed value. The calculations are based on FNP instrumentation and test methods and include uncertainty allowances. The evaluations and analyses results [a small change] demonstrate applicable acceptance criteria are met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief (Acting): Maitri Banerjee.

Southern Nuclear Operating Company, Inc, Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: December 8, 2000.

Description of amendment request: The proposed amendments would either delete or modify existing license conditions from the Unit 1 and Unit 2 Operating Licenses, which have been completed or are otherwise no longer in effect. These activities have now been completed, and the license conditions are either obsolete or no longer needed.

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 amendment deletes license conditions which are completed or are otherwise obsolete. As such, the change is strictly administrative. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment deals with operating license reporting conditions and has no effect on the type of accidents that have been considered at Plant Farley. Therefore, this change does 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 requirements associated with the deleted license conditions have been completed; the conditions are therefore obsolete. Removing these conditions from the license is an administrative and editorial activity. 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief (Acting): M. Banerjee.

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

Date of application request: January 18, 2001 (ULNRC–04371).

Description of amendment request: The proposed amendment deletes Section 5.5.3, ''Post Accident Sampling,'' from the administrative controls section of the Technical Specifications (TS). The proposed amendment deletes requirements from the TS (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated January 18, 2001.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve as a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: September 14, 2000.

Brief description of amendments: The amendments add two analytical methods to the list of approved core operating limit analytical methods in Technical Specification 5.6.5.b.

Date of issuance: February 8, 2001.

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

Amendment Nos.: 241 and 215.

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

Date of initial notice in ***Federal Register:*** October 18, 2000 (65 FR 62383).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated February 8, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 22, 1999, as supplemented on September 11, 2000.

Brief description of amendment: The amendment revises Technical Specification Sections 4.5.D, ''Containment Air Filtration System,'' 4.5.E, ''Control Room Air Filtration System,'' 4.5.F, ''Fuel Storage Building Air Filtration System,'' and 4.5.G, ''Post-Accident Containment Venting System,'' to address the testing requirements in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The laboratory testing of the engineered safeguards features ventilation system charcoal samples will meet the requirements of the American Society for Testing and Materials Standard D3803–1989.

Date of issuance: February 21, 2001.

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

Amendment No.: 215.

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

Date of initial notice in ***Federal Register:*** November 15, 2000 (65 FR 69059).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 21, 2001.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: October 30, 2000.

Brief description of amendment: The amendment revises Surveillance Requirement 3.6.1.3.8 to allow a representative sample of reactor instrument line excess flow check valves (EFCVs) to be tested every 24 months such that each reactor instrument EFCV will be tested at least once every 10 years. The amendment also limits the surveillance requirement to only the reactor instrument line EFCVs.

Date of issuance: February 20, 2001.

Effective date: February 20, 2001, and shall be implemented within 30 days from the date of issuance.

Amendment No.: 170.

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

Date of initial notice in ***Federal Register:*** November 29, 2000 (65 FR 71135).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 20, 2001.

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: November 22, 1999, as supplemented on November 21, 2000.

Brief description of amendment: This amendment approves changes related to Technical Specification (TS) Sections 3.7.B.1 and 3.7.B.2, ''Containment Systems.'' TS Section 5.0, ''Administrative Controls,'' was also modified to reflect the addition of an omitted page from a previous amendment.

Date of issuance: February 13, 2001.

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

Amendment No.: 187.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17913).

The November 21, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: February 15, 2000, as supplemented on July 26, 2000. The July 26, 2000, letter provided clarifying information that did not change the scope of the February 15, 2000, application or the initial proposed no significant hazards consideration determination.

Brief description of amendments: The amendments allow the use of the Westinghouse core monitoring system know as Best Estimate Analyzer for Core Operations Nuclear.

Date of issuance: February 13, 2001.

Effective date: February 13, 2001.

Amendment Nos.: 116, 116, 110, and 110.

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

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17909).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 26, 2000.

Brief description of amendment: The amendment changes the Technical Specifications to (1) allow reactor vessel hydrostatic tests, leakage tests, scram time tests and excess flow check valve tests be performed; (2) require containment building integrity be maintained; and (3) establish a limit and a surveillance requirement on reactor coolant radioactive iodine activity, when coolant temperature is above 215 °F, the reactor is not critical, and primary containment integrity has not been established.

Date of issuance: February 20, 2001.

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

Amendment No.: 170.

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

Date of initial notice in ***Federal Register:*** November 1, 2000 (65 FR 65344).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated February 20, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: November 10, 2000.

Brief description of amendment: The amendment revised several sections of the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs). These sections include administrative changes, Table 4.1–1, and Sections 1.0, 6.4, and 6.10.

Date of issuance: February 12, 2001.

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

Amendment No.: 151.

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

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77923).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 12, 2001.

No significant hazards consideration comments received: No.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of application for amendment: August 5, 1999, as supplemented by letters dated November 23, 1999, December 27, 1999, May 4, 2000, October 19, 2000, and November 22, 2000.

Brief description of amendment: The amendment revised the Facility Operating (Possession Only) License to annotate approval of the Trojan Nuclear Plant License Termination Plan.

Date of issuance: February 12, 2001.

Effective date: February 12, 2001, and shall be implemented within 30 days of the effective date.

Amendment No.: 206.

Facility Operating License No. NPF–1: The amendment changes the Facility Operating (Possession Only) License.

Date of initial notice in ***Federal Register:*** December 29, 1999 (64 FR 73083). The November 23, 1999, December 27, 1999, May 4, 2000, October 19, 2000, and November 22, 2000, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 12, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 4, 2000 (TS 99–20).

Brief description of amendments: Deletes Sequoyah License Condition for Shift Technical Advisor and revises Technical Specifications (TSs) that specify shift manning requirements.

Date of issuance: February 16, 2001.

Effective date: February 16, 2001.

Amendment Nos.: 266 and 257.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the Operating Licenses and TSs.

Date of initial notice in ***Federal Register:*** September 6, 2000 (65 FR 54088).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 21, 2000 (ULNRC–04346).

Brief description of amendment: The amendment changes Table 3.3.2–1, ''Engineered Safety Feature Actuation System [ESFAS] Instrumentation,'' of the Technical Specifications. The change adds Surveillance Requirement (SR) 3.3.2.10 for the following two ESFAS instrumentation in the table: item 6.f, loss of offsite power, and item 6.h, auxiliary feedwater pump suction transfer on suction pressure—low.

Date of issuance: February 12, 2001.

Effective date: February 12, 2001, and shall be implemented prior to entering Mode 3 from Mode 4 during the startup from Refuel Outage 11, including the revision of the FSAR to reflect the ESFAS response times in accordance with the application.

Amendment No.: 141.

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

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81931).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 12, 2001.

No significant hazards consideration comments received: No.

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

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

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

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: February 1, 2001.

Brief description of amendment request: The amendment would remove the inservice inspection requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code from the Monticello Technical Specifications and relocates them to a licensee-controlled program.

Date of publication of individual notice in ***Federal Register:*** February 15, 2001 (66 FR 10535).

Expiration date of individual notice: March 1, 2001.

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

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

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

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By April 6, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, and to the attorney for the licensee.

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

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: February 14, 2001, as supplemented February 16 and 19, 2001. The February

16 and 19, 2001, letters provided additional clarifying information which did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice (Harrisburg, PA, Patriot News, February 18–20, 2001).

Brief description of amendment: The amendment allows a one-time exception to the system configuration and maintenance requirements in Technical Specification (TS) 3.3.2 related to the nuclear service river water (NR) system at TMI–1, in order to allow an up to 14-day repair of a leaking underground concrete pipe. The requirements of TS 3.3.1.4 to have two NR pumps OPERABLE are unchanged. During the 14-day repair period, the NR pumps flow will be realigned to pass through a portion of the nonseismic secondary services river water system.

Date of issuance: February 23, 2001.

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

Amendment No.: 229.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

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

The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on February 23, 2001. The notice was published in the Harrisburg, PA, Patriot News, from February 18 through February 20, 2001.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Pennsylvania, and final no significant hazards consideration determination are contained in a Safety Evaluation dated February 23, 2001.

Attorney for licensee: Edward J. Cullen, Jr., Esquire, PECO Energy Company, 2301 Market Street (S23–1), Philadelphia, PA 19103.

NRC Section Chief: Marsha Gamberoni.

Dated at Rockville, Maryland this 27th day of February 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–5216 Filed 3–6–01; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Proposed Collection; Comment Request for Review of a Revised Information Collection: RI 94–7

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Public Law 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) intends to submit to the Office of Management and Budget a request for review of a revised information collection. RI 94–7, Death Benefit Payment Rollover Election for Federal Employees Retirement System (FERS), provides FERS surviving spouses and former spouses with the means to elect payment of the FERS rollover-eligible benefits directly or to an Individual Retirement Account.

Comments are particularly invited on: whether this information is necessary for the proper performance of functions of OPM, and whether it will have practical utility; whether our estimate of the public burden of this collection of information is accurate, and based on valid assumptions and methodology; and ways in which we can minimize the burden of the collection of information on those who are to respond, through the use of appropriate technological collection techniques or other forms of information technology.

Approximately 700 RI 94–7 forms will be completed annually. We estimate it takes approximately 60 minutes to complete the form. The annual estimated burden is 700 hours.

For copies of this proposal, contact Mary Beth Smith-Toomey on (202) 606–8358, or E-mail to mbtoomey@opm.gov

DATES: Comments on this proposal should be received on or before May 7, 2001.

ADDRESSES: Send or deliver comments to: John C. Crawford, Chief, FERS Division, Retirement and Insurance Service, U.S. Office of Personnel Management, 1900 E Street, NW, Room 3313, Washington, DC 20415.

FOR INFORMATION REGARDING ADMINISTRATIVE COORDINATION CONTACT: Donna G. Lease, Team Leader, Forms Analysis and Design, Budget and Administrative Services Division, (202) 606–0623.

U.S. Office of Personnel Management.

Steven R. Cohen,

Acting Director.

[FR Doc. 01–5517 Filed 3–6–01; 8:45 am]

BILLING CODE 6325–50–U

SECURITIES AND EXCHANGE COMMISSION

[Release No. IC–24881; 812–12266]

ING Pilgrim Investments, LLC, et al.; Notice of Application

February 28, 2001.

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

ACTION: Notice of application for an order under section 6(c) of the Investment Company Act of 1940 (''Act'') for an exemption from sections 18(c) and 18(i) of the Act, under sections 6(c) and 23(c)(3) of the Act for an exemption from rule 23c-3 under the Act, and pursuant to section 17(d) of the Act and rule 17d-1 under the Act.

SUMMARY OF APPLICATION: Applicants request on order to permit certain registered closed-end management investment companies to issue multiple classes of shares and to impose asset-based distribution fees and early withdrawal charges.

APPLICANTS: Pilgrim Senior Income Fund (''Fund''), ING Pilgrim Investments, LLC (''Investment Adviser''), and ING Pilgrim Securities, Inc. (''ING Pilgrim Securities'').

FILING DATES: The application was filed on September 25, 2000 and amended on February 28, 2001.

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 applicants with a copy of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on March 26, 2001, and should be accompanied by proof of service on applicants, 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 Fifth Street, NW, Washington, DC 20549–0609. Applicants, 7337 East Doubletree Ranch Road, Scottsdale, Arizona, 85258.