January 25, 1988

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DOCKET NO(S). 50-321
50-366

Mr. James P. O'Reilly
Senior Vice President-
Nuclear Operations
Georgia Power Company
P.O. BOx 4548
Atlanta, Georgia 30302

SUBJECT: E. I. Hatch Nuclear Plant, Units 1 and 2

The following documents concerning our review of the subject facility are transmitted for your information.

- [ ] Notice of Receipt of Application, dated ____________.
- [ ] Draft/Final Environmental Statement, dated __________.
- [ ] Notice of Availability of Draft/Final Environmental Statement, dated ___________.
- [ ] Safety Evaluation Report, or Supplement No. ____________________ dated __________.
- [ ] Environmental Assessment and Finding of No Significant Impact, dated ___________.
- [ ] Notice of Consideration of Issuance of Facility Operating License or Amendment to Facility Operating License, dated __________.
- [x] Bi-Weekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations, dated 1/13/88 [see page(s)] 827 and 831.
- [ ] Exemption, dated __________.
- [ ] Construction Permit No. CPPR-__________, Amendment No. ___________ dated __________.
- [ ] Facility Operating License No. __________, Amendment No. ___________ dated __________.
- [ ] Order Extending Construction Completion Date, dated __________.
- [ ] Monthly Operating Report for ________________ transmitted by letter dated __________.
- [ ] Annual/Semi-Annual Report- ____ transmitted by letter dated _________.

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Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

OFFICE	PD#23/DPR-I/II						
SURNAME	R. Rood	L. Crocker					
DATE	1/25/88	1/25/88					

NRC FORM 318 (10/80) NRCM 0240

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NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law (P.L.) 97-415, the Nuclear Regulatory Commission (the Commission) is publishing this regular biweekly notice. P.L. 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 December 21, 1987 through December 31, 1987. The last biweekly notice was published on December 30, 1987 (52 FR 49217).

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR 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 amendments 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. The Commission will not normally make a final determination unless it receives a request for a hearing.

Written comments may be submitted by mail to the Rules and Procedures Branch, Division of Rules and Records, Office of Administration and Resource Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 4000, Maryland National Bank Building, 7735 Old Georgetown Road, Bethesda, Maryland from 8:15 a.m. to 5:00 p.m. Copies of written comments received may be examined at the NRC Public Document Room, 1717 H Street, NW., Washington, DC. The filing of requests for hearing and petitions for leave to intervene is discussed below.

By February 12, 1988 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 petition for leave to intervene. Requests for a hearing and petitions 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. 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 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 fifteen (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 fifteen (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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. 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 involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

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 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.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW., Washington, DC, by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to (*Project Director*): petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel-Bethesda, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding 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, 1717 H Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

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

Date of amendment request: January 20, 1987

Description of amendment request: The following proposed Technical Specification (TS) changes are in response to the BG&E application dated January 20, 1987. The proposed TS changes:

(1) Modify the Unit 1 TS Limiting Condition For Operation (LCO) 3.3.3.2 for incore detectors by placing additional restrictions upon operability

above those that were required for operation during the previous cycle (Cycle 8).

(2) Change the surveillance periods of the Unit 1 and 2 TS Surveillance Requirements (SRs) 4.1.3.4.c (demonstration of full length control element assembly (CEA) drop time) and 4.3.3.2.b (incore detector channel calibration) from at least once per 18 months to at least once per refueling interval, where a refueling interval shall be defined as 24 months.

(3) Modify the Units 1 and 2 TS SR 4.7.11.1.1.f.3, for cycling fire suppression water system flow path valves that are not testable during plant operation, and 4.7.11.4.b, for the inspection, reracking and replacement of degraded coupling gaskets for fire hoses inside containment, by extending their associated surveillance intervals from at least once every 18 months to at least once per refueling interval (24 months), and

(4) Renumber the Units 1 and 2 TS SR 4.7.11.1.1.f.3 as 4.7.11.1.1.g.2 and TS SR 4.7.11.1.1.g as 4.7.11.1.1.g.1 and change the Units 1 and 2 TS SRs 4.7.11.1.1.g (fire suppression system flow test), 4.7.11.2.b and c (spray and sprinkler system functional tests), and 4.7.11.4.c (containment fire hose stations operability and hydrostatic tests) by making administrative changes and more restrictive changes to the surveillance requirements.

Basis for proposed no significant hazards consideration determination: Change No. 1 proposes to modify the Unit 1 TS LCO 3.3.3.2 for incore detector operability by making its provisions more restrictive than those required for Unit 1 Cycle 8 operation. During startup for Unit 1 Cycle 8, an unexpectedly large number of incore detector strings failed thereby placing the Unit close to its operability limits. To provide increased operational flexibility for Unit 1 during Cycle 8 operations, the requirements of TS LCO 3.3.3.2 were relaxed for one cycle only. In order to restore LCO 3.3.3.2 to its pre-cycle 8 requirements, the following modifications are proposed:

(1) LCO 3.3.3.2.a would require at least eight operable symmetric incore detector segment groups, with at least two of these detector segment groups at each of the four axial elevations containing incore detectors, to have sufficient operable detector segments to compute at least two azimuthal power tilt valves at each of these four axial elevations. During Cycle 8, eight symmetric incore detector segment groups of no specified elevation were required with sufficient operable detector segments to compute at least two azimuthal power tilt valves at three of the four axial elevations.

(2) LCO 3.3.3.2.b would require that at least 75% of all incore detector segments be operable for recalibration of the excore neutron flux detection system rather than the 50% required during Cycle 8.

(3) LCO 3.3.3.2.c would require, for monitoring the unrodded planar radial peaking factor, the unrodded integrated radial peaking factor, or the linear heat rate, that at least 75% of all incore detector locations be operable rather than the 50% required during Cycle 8.

On March 6, 1986, the NRC published guidance in the **Federal Register** (51 FR 7751) concerning examples of amendments that are not likely to involve a significant hazards consideration.

One of the examples, (ii) was "a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications." This proposal is one such change as the proposed TS LCO modifications make the operability requirements for incore detectors more restrictive than those currently specified for Cycle 8.

Based upon the above, the NRC staff agrees with the licensee's evaluation and proposes to determine that the proposed changes to TS LCO 3.3.3.2 involve no significant hazards considerations.

Change No. 2 proposes to change the surveillance periods from 18 to 24 months for the Units 1 and 2 TS surveillance requirements for demonstrating full length CEA drop time (TS 4.1.3.4.c) and for performing incore detector channel calibrations (TS 4.3.3.2.b).

The current surveillance period for these tests is 18 months which corresponds to the current refueling cycle. The extension in the surveillance interval to 24 months is requested to facilitate a 24-month operating cycle.

The licensee evaluated the proposed change against the standards in 10 CFR 50.92 and has determined that the amendment would not:

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

The licensee has proposed a six-month extension in the surveillance period of TS 4.1.3.4.c for performing CEA drop time test. Control element assembly drop time is required to be less than or equal to 3.1 seconds. The CEA drop time is measured from the time that electrical power is interrupted to a fully withdrawn CEA to the time required for the CEA to be at its 90% insertion position. This drop time testing is performed at a reactor coolant system average temperature greater than or equal to 515° F and with all four reactor coolant pumps operating. These conditions are representative of reactor conditions for reactor trips from operating conditions. The purpose of the CEA drop time testing is to ensure that scram insertion times are consistent with those used in the safety analyses. Factors which could adversely affect the CEA drop times when the surveillance interval is increased are (1) changes in component clearances, (2) changes in the physical configuration of the CEA or guide tubes, and (3) the buildup of corrosion products and suspended material in the coolant system that could interfere with CEA motion. Changes to component clearances and changes in the physical configuration of the CEA or guide tubes are more likely to occur when the reactor vessel head is removed and when maintenance is performed on the CEAs (including replacement) and that portion of the drive system directly interfacing with a fuel assembly. For these two factors, Surveillance Requirements 4.1.3.4.a and 4.1.3.4.b are applicable and not affected by the proposed change in the testing interval of Surveillance Requirement 4.1.3.4.c. Buildup of corrosion products and suspended material in the coolant system are minimized by coolant chemistry requirements and other controls on the reactor coolant system. In addition, each CEA is exercised at least once per 31 days in accordance with Surveillance Requirement 4.1.3.1.2. This testing should detect sticking CEAs and mitigate the proposed 6-month extension in the surveillance interval of TS 4.1.3.4.c for demonstrating CEA drop time. Furthermore, each planned or unplanned reactor trip that may occur during the extended 24-month operating cycle would provide additional information on CEA drop times and operability, thus, indicating any problems developing with regards to CEA drop time.

To determine the time dependency of CEA drop time with respect to the length of the operating cycle, CEA drop time measurements from 15 hot functional test data/sets were analyzed. Eight sets of measurements were taken from Unit 1 and seven from Unit 2. The average CEA drop time for standard fuel assemblies was approximately 2.3 seconds. The maximum standard deviation for drop times in any fuel cycle was 0.094 seconds. The 15 sets of test data included data from both 12-month and 18-month fuel cycles. Thus, this data indicates that no increase in drop time trend was observed due to either lengthening the operating cycles or to increased periods between surveillance testing from 12 to 18 months.

The licensee's analysis of previous fuel cycle CEA drop time measurements, which showed no adverse affects when shifting from a 12-month to an 18-month cycle, as well as the other surveillance requirements that are performed to determine CEA drop time, indicate that the CEA drop time should not be appreciably affected by the proposed 6-month extension of the surveillance period of the TS 4.1.3.4.c to 24 months. Hence, the probability or consequences of previously evaluated accidents would not be significantly increased.

Also, the licensee has proposed a 6-month extension in the surveillance interval of TS 4.3.3.2.b for performing incore detector channel calibrations. The incore detector channel calibration excludes the neutron detectors but includes all electronic components. The channel calibration consists of two parts: (1) a resistance check of the cable from the computer termination to the reactor core, and (2) a check of the ability of the computer to read a known voltage level. The resistance check verifies cable integrity. A review of resistance checks performed since the initial startups of Calvert Cliffs Units 1 and 2 has been conducted. No evidence of cable degradation was found. However, all of the in-containment cable is being replaced with environmentally qualified cable. The design specification for the new cable will ensure that it is at least as reliable as the cable it replaces. The second part of the channel calibration checks the computer's ability to read a known voltage level. Three known signals are input into the computer: (1) a short circuit, (2) a 150 millivolt signal, and (3) a 250 millivolt signal. Proper computer readings are verified for each test with the voltages being between ± 2 millivolts. Other checks to verify proper computer operation are also performed and include CRT and alarm printer verification.

Test data from the initial units' startups to the present has been reviewed to determine if performance changed in an adverse manner over time and with the shift from a 12-month to an 18-month operating cycle. The licensee noted no adverse trends and found that all tests had been consistently satisfactory.

In addition, performance of the power distribution TS Surveillance Requirements 4.2.2.1.2 and 4.2.3.2, which are at least once per 31 mode 1 days, provides further assurance of the operability of the incore detection system. The licensee states that if the incore detector system was to be inoperable, other methods are employed to carry out its monitoring and calibration functions.

The licensee's analysis of previous fuel cycle incore detection system calibration data, which showed no adverse trends when shifting from a 12-month to an 18-month cycle, as well as the power distribution surveillance requirements that are imposed at least once every 31 days of mode 1 operation, indicate that the operability of the incore detectors should not be appreciably affected by the proposed 6-month extension to 24 months of the surveillance interval of TS 4.3.3.2.b for performing incore detector channel calibrations. Hence, the probability or consequences of previously evaluated accidents would not be significantly increased.

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

The proposed changes only extend the surveillance intervals for CEA drop time testing and incore detector calibrations. This proposal does not change any system design or facility operation; therefore, it does not create the possibility of a new or different kind of accident from any previously evaluated.

(iii) Involve a significant reduction in a margin of safety...

The margins of safety that could be potentially affected by these changes included the margin from reactor coolant system overpressurization and the margins from peak centerline temperature (PCT) and from the depature from nucleate boiling (DNB), due to a possible decrease in the negative reactivity insertion rate on a reactor trip and from inaccurate flux monitioring due to degradation in the incore detectors.

However, a review of plant surveillance history shows that: (1) both of these systems (CEAs and incore detectors) have been extremely reliable, and (2) the surveillance results of both systems have routinely yielded excellent results that were independent of the time between surveillances (cycle length). In addition, in both cases there exists other TS surveillance requirements that monitor CEA and incore detector performance and would most likely indicate any ongoing degradation in either system, thus mitigating any potential hazards presented by extending the surveillances intervals. Therefore, this change does not involve any significant reduction in a margin of safety.

Based upon the above, the NRC staff proposes to determine that the proposed changes to TS Surveillance Requirements 4.1.3.4.c and 4.3.3.2.b involve no significant hazards considerations.

Change No. 3 proposes to modify the Units 1 and 2 TS SRs 4.7.11.1.1.f.3, for cycling fire suppression water system flow path valves that are not testable during plant operation, and 4.7.11.4.b, for the inspection, re-racking and replacement of degraded coupling gaskets for fire hoses inside containment, by extending their associated surveillance intervals from at least once every 18 months to at least once per refueling interval (24 months).

The interval for these surveillances is 18 months which corresponds to the current refueling cycle. The extension of the surveillance interval to 24 months is requested to facilitate a 24-month operating cycle.

The licensee evaluated the proposed change against the standards in 10 CFR 50.92 and has determined that the amendment would not:

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

The proposal to modify TS Surveillance Requirement 4.7.11.1.1.f.3 affects only two fire suppression water system valves inside containment. LCO 3.7.11.1.c at all times requires an operable fire suppression water system flow path that takes a suction from the water storage tanks and transfers the water through the distribution system up to the first valve before the water flow alarm device on each sprinkler, hose standpipe or spray system riser. All valves in this flow path can be tested during unit operation with the exception of the two valves inside containment (the motor operated containment isolation valve and a manual block valve). TS Surveillance Requirement 4.7.11.1.1.f.3 requires these two valves to be tested by cycling and verifying flow. The licensee's results from a review of plant history indicate that there has never been a failure of either valve to perform adequately. The licensee further states that there is no evidence that a 6-month extension in this surveillance interval between valve cycles would adversely impact valve operation. Hence, the probability or consequence of previously evaluated accidents would not be significantly increased by the proposed 6-month extension of the surveillance interval of TS 4.7.11.1.1.f.3.

The proposed modification of TS Surveillance Requirement 4.7.11.4.b would affect only the inspection and reracking of fire hoses inside containment. A review of previously conducted containment fire hose inspections revealed no failures of the fire hoses. The licensee states that these results were expected as it has been a licensee policy to replace all fire hoses inside containment on a three-year frequency. The licensee intends, for the 24-month operating cycle, to hydrostatically test or replace all containment fire hoses every two years.

Furthermore, test results have shown that the hose coupling gasket material has not degraded significantly over the three-year interval between hose replacements. Finally, during hose inspection, there has never been evidence of hose mildew, rot or similar damage due to chemicals, abrasion, moisture or normal wear. Thus, it is unlikely that the containment fire hoses would experience any significant degredation over the proposed 6-month surveillance interval extension. Hence, the probability or consequences of the proposed change to TS 4.7.11.4.b would not significantly increase the probability or consequences of any previously evaluated accidents.

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

These proposed changes do not create the possibility of any new or different accidents as no plant modifications or changes in system operation or surveillance testing, other than test interval, shall be made.

(iii) Involve a significant reduction in a margin of safety...

Extending the surveillance interval for these two tests does not involve a significant reduction in any margin of safety. Efforts were made to extend the surveillance interval of only those tests that could not be performed during unit operation (i.e., testing and inspecting fire hoses and fire suppression water system valves inside containment). These containment fire protection components are generally inaccessible during unit operation, and so, will be tested during refueling outages. However, the likelihood of a fire inside containment during unit operation is much smaller than during outage work periods. Thus, the likelihood of a fire occurring inside containment that would damage safety and safety-related systems will not be significantly increased by this proposed 6-month test interval extension. Therefore, the margins of safety provided by these safety and safety-related systems will not be significantly reduced.

Based upon the above, the NRC staff proposes to determine that the proposed changes to TS SRs 4.7.11.1.1.f.3 and 4.7.11.4.b involve no significant hazards considerations.

Change No. 4 proposes to renumber the Units 1 and 2 TS SR 4.7.11.1.1.f.3 as 4.7.11.1.1.g.2 and TS SR 4.7.11.1.1.g as 4.7.11.1.1.g.1 and to modify the Units 1 and 2 TS SRs 4.7.11.1.1.g, 4.7.11.2.b & c and 4.7.11.4.c by making administrative or more restrictive changes to the current surveillance requirements. The proposed restrictive changes to the surveillance requirements are as follows:

(1) the surveillance interval for performing a fire suppression water system flow test in accordance with TS 4.7.11.1.g would be changed to "at least once per refueling interval" (24 months) from the currently required "at least once per 3 years."

(2) the spray and sprinkler system cycling test of each flow path valve would be conducted at least every 12 months. Currently, only testable valves are required to be cycled at least every 12 months by TS 4.7.11.2.b, whereas TS 4.7.11.2.c.1.b requires the cycling of those not testable during plant operation at least every 18 months. All of these valves, however, are testable during plant operation, making TS 4.7.11.2.c.1.b superfluous. Consequently, the licensee has proposed deletion of TS 4.7.11.2.c.1.b and of the word "testable" from the phrase "by cycling each testable valve" in TS 4.7.11.2.b.

(3) fire hose station valve operability and hose hydrostatic tests currently are required by TS 4.7.11.4.c to be performed at least once per 3 years. The licensee has proposed that these tests on fire hose stations inside containment be required to be performed at least once during refueling interval (24 months).

On March 6, 1986, the NRC published guidance in the **Federal Register** (51 FR 7751) concerning examples of amendments that are not likely to involve a significant hazard consideration.

Two of the examples were "(i) A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature" and "(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications, e.g., a more stringent surveillance requirement." These proposals are such administrative and more restrictive changes as they simplify the TS to better reflect plant conditions and also, require surveillances to be performed more frequently.

Based upon the above, the NRC staff agrees with the licensee's evaluation and proposes to determine that the proposed changes to TS 4.7.11.1.1.f.3, 4.7.11.1.1.g, 4.7.11.2.b, & c and 4.7.11.4.c involve no significant hazards considerations.

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

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

NRC Project Director: Robert A. Capra, Director

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

Date of application for amendment: November 6, 1987 as supplemented by December 16, 1987

Description of amendment request: The proposed amendment would revise Tables 3.2-1 and 4.2-1 of the Quad Cities, Units 1 and 2, Technical Specifications (TS) for High Pressure Core Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems Steam Line High Flow Indication Instrumentation. More specifically, a TS amendment was requested that would: (1) revise the number of operable or tripped HPCI and RCIC steam line high flow indication instrument channels from a minimum of four (4) channels to two (2) channels; this will correct a discrepancy that has existed since the original TS were issued, by making the number of channels consistent with the original design basis and actual plant configuration, and (2) revise the HPCI and RCIC high steam flow time delay setting of $3 \leq t \leq 10$ seconds to a more conservative setting of $3 \leq t \leq 9$ seconds; this change was recommended by the Commonwealth Edison Company (CECo, the licensee) Engineering Department based upon General Electric (GE) Company analysis.

Additionally, the TS amendment would correct a typographical error in the associated surveillance requirement bases. Current TS for Units 1 and 2 identify the high steam flow instruments as 1-2389 A thru D and 2-2389 A thru D, while the correct designations are 1-2352, 1-2353, 2-2352 and 2-2353. The low pressure instruments are listed as 1-2352, 1-2353, 2-2352 and 2-2353 in the Units 1 and 2 TS, while the correct designations are 1-2389 A thru D and 2-2389 A thru D. Instrument numbers for the high steam flow instrumentation were actually the designations for the low pressure instrumentation while the instrument numbers for the low pressure instrumentation, are actually the designations for the HPCI high steam flow instruments. Revising these instrument designations is considered to be an administrative change.

The application for amendment was originally noticed in the **Federal Register** (52 FR 45884) on December 2, 1987. CECo supplemented their initial submittal on December 16, 1987 to incorporate the proposed time delay setting into applicable surveillance requirements of Table 4.2-1. This revised time delay setting for TS table 4.2-1 had been inadvertently omitted from the November 6, 1987 application for amendment.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequence 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. In accordance with 10 CFR 50.91(a), the licensee has provided the following analysis in their amendment application addressing these three standards.

CECo has analyzed this proposed amendment and determined 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 because:

a. Previously evaluated accidents were based on two channels for the RCIC and HPCI steam line flow indications rather than four; this means that the evaluations were based on conditions that actually exist in the plant, not the number of channels found in the current Technical Specifications. Plant operations and accident analyses are not changed.

b. The proposed time delay setting is lower than the setting which currently exists. Operating with a maximum time delay setting of nine seconds is more conservative that the previously approved ten second value.

c. Changing instrument designation to correct typographical errors are considered to be an administrative change and has no effect upon previously evaluated accident scenarios.

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

a. The number of HPCI and RCIC instrument channels are corrected to reflect the number of channels that actually exist, and upon which the original system design was based. The manner in which the plant has been, or will be operated does not change. Additionally, operating with a minimum number of two tripped or operable HPCI or RCIC high flow instrument channels is more conservative than with four channels.

b. The new time delay setting is more conservative than the value that currently exists in the Quad Cities TS.

c. Correction of typographical errors are considered to be administrative in nature and have no effect on plant operation.

3. Involve a significant reduction in the margin of safety because:

a. The number of HPCI and RCIC instrument channels are corrected to reflect actual plant configuration and original design. There are no changes being made to hardware. The proposed amendment does not reduce the margin of safety since the minimum number of operable or tripped channels will be more conservative.

b. The new maximum time delay setting will be more conservative than the value currently in TS.

c. Correction of typographical errors involve the designation for HPCI instrumentation only, safety margins are unaffected.

The Commission has reviewed the licensee's TS amendment request and concurs with their analysis for no significant hazards consideration determination. Accordingly, the Commission proposes to determine the aforementioned amendment request does not involve a significant hazards consideration.

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

Attorney to licensee: Michael I. Miller, Esq. of Isham, Lincoin, & Beale at Three First National Plaza, Suite 5200, Chicago, Illinois 60602.

NRC Project Director: Daniel R. Muller

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

Date of application for amendment: December 22, 1987

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 3.2.D.3 (refueling floor radiation monitor setpoint) and associated bases, and TS 6.2.C.1 (review responsibilities for changes to procedures).

Current TS establish a trip setpoint of 100 mR/hr for refueling floor radiation monitors. Commonwealth Edison Company (CECo, the licensee) has proposed revising this setpoint to "less than or equal to 100mR/hr" in order to prevent possible inadvertent trips during instrument calibration. These radiation monitors are calibrated to 100 mR/hr which does not allow for normal instrument setpoint drift if the TS trip setpoint is also at 100 mR/hr.

Instead of allowing review and approval responsibilities to be split up among various subject areas as required by present TS, the proposed amendment would prescribe that all procedure changes "shall be reviewed and approved by the Technical Staff Supervisor, the Assistant Superintendent, and department head...". Altering review and approval responsibilities for changes to procedures (identified in TS Section 6.A and 6.B), in this fashion, should increase consistency and improve uniformity for all areas of plant activities. Furthermore, this proposed amendment will also elevate review responsibility to a higher level of management.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequence 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. In accordance with 10 CFR 50.91(a), the licensee has provided the following analysis in their amendment application addressing these three standards.

CECo has analyzed this proposed amendment and determined 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 because the proposed amendment merely establishes an upper bound limit for the refuel floor radiation monitors consistent with what currently exists in the Tech Specs. This is considered to be a change in the conservative direction and will not effect system design or safety function.

The proposed amendment also raises the level of reviews for procedure changes to the Assistant Superintendent level for all procedures identified in Section 6.2.A. and 6.2.B. of the TS. This change results in a higher level of approval for changes to procedures than is currently provided in the TS. This change is considered to be administrative in nature and should improve the quality of plant procedures used to operate the station.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed amendment does not exceed the existing setpoint for refuel floor radiation monitors, but rather makes 100mR/hr the upper bounding value. There are no hardware

changes, nor are there any new modes of operation associated with this amendment.

Revised review and approval responsibilities for procedures changes would be an administrative change. No new equipment or modes of operation have been introduced as a result of this TS revision. Revising the authorization level for procedure changes to a higher level does not introduce any new equipment or modes of operation at Quad Cities Station.

3. Involve a significant reduction in the margin of safety because the setpoint of 100mR/hr is not being changed to a different value, but rather is becoming an upper bounding value for the refuel floor radiation monitors. This will prevent inadvertent trips which may occur because of normal instrument drift and unnecessary system challenges. Any deviation from the 100mR/hr setpoint allowed by the proposed TS change would result in an increased margin (i.e. radiation level setpoint is only allowed to be lowered).

Section 6 revisions are considered to be administrative in nature. This TS revision being proposed does not result in hardware modifications that would effect the way plant systems are being operated.

The Commission has reviewed the licensee's TS amendment request and concurs with their analysis for no significant hazards consideration determination. Accordingly, the Commission proposes to determine the aforementioned amendment request does not involve a significant hazards consideration.

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

Attorney to licensee: Michael I. Miller, Esq. of Isham, Lincoln, & Beale at Three First National Plaza, Suite 5200, Chicago, Illinois 60602.

NRC Project Director: Daniel R. Muller

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

Date of amendment request: May 26, 1987.

Description of amendment request: The proposed change would revise the license condition for the receipt, possession and use of byproduct, source and special nuclear material in accordance with a standard, generalized format that allows flexibility in amounts of such material in support of reactor operation.

Basis for proposed no significant hazards consideration determination: The licensee has evaluated this proposed amendment for determining whether or not it involves a significant hazards consideration as follows:

The control of byproduct, source or special nuclear material sources exceeding 100 millicuries is by approved Radiological Services Department procedures which contain information described in Regulatory Guide 1.70.

The ability to handle sources has been demonstrated at Palisades since the Provisional Operating License was issued. Personnel qualifications, facilities, and equipment and procedures for handling have also been established. Surveillance leak testing to determine source leakage was incorporated into the Technical Specification, Section 6.21, approved in Amendment No. 98.

The amount of reactor fuel which can be received, possessed, and used may vary from the present license limit but will be limited by available storage and amounts required for operation.

The changes do not involve a significant hazards consideration at Palisades as this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change revises the license conditions for the amount of special nuclear material, source material, and byproduct material in accordance with the NRC's letter of January 24, 1975, with some modification. Provisions to ensure reactor fuel is limited to amounts compatible with the present possession amounts are controlled by the amount of storage space and fuel necessary for reactor operation as described in the FSAR.

The sources will be adequately leak tested, stored and used and records will be maintained as required by the Technical Specification, Section 6.21.

(2) Create the possibility of a new or different kind of accident from any previously analyzed. Appropriate controls for receipt, handling and storage of the special nuclear material, byproduct material and source material are in place and remain unchanged as a result of this request to ensure no new or different accident will be created.

(3) Involve a significant reduction in a margin of safety. The controls over the receipt, handling and storage remain unchanged as a result of this request. These controls will ensure no safety margin is reduced.

The Palisades Plant Review Committee has reviewed this Technical Specification Change Request and has determined that this change involves no significant hazards consideration.

The Commission's staff has reviewed the licensee's evaluation and agrees. The staff therefore proposes to determine that this proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: Judd L. Bacon, Esq., Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: Martin J. Virgilio.

General Public Utilities Nuclear Corporation, Docket No. 50-320, Three Mile Island Nuclear Station Unit 2 (TMI-2), Dauphin County, Pennsylvania

Date of amendment request: April 23, 1987, revised October 26, 1987, November 9, 1987, and December 4, 1987.

Description of amendment request: The proposed amendment would revise TMI-2 Operating License No. DPR-73 by modifying Appendix A Technical Specifications Sections 2 - Safety Limits, 3 - Limiting Conditions for Operation, 3/4 - Basis for Limiting Conditions for Operations and Surveillance Requirements, and 6 - Administrative Controls. Additionally, the proposal would amend the Index. The proposed amendment would extensively revise the TMI-2 Technical Specifications to align license requirements appropriate to current, as well as future, plant conditions through the remainder of the current cleanup operations. At the end of the current cleanup operations the licensee plans to place the facility into a post-defueling monitored storage condition (PDMS). The proposed amendment allows for the transition from the current defueling phase through the completion of defueling and offsite fuel shipment by the incorporation of technical specifications that are applicable during specific phases or modes of the cleanup. Certain technical specifications are retained during the entire transition period while others are phased out or modified as the cleanup progresses. Phase-out of specific requirements would be dependent on the status of the cleanup as defined by the facility mode. Three cleanup modes are proposed:

Mode 1 - The current condition, during which defueling and other major tasks are in progress.

Mode 2 - The period subsequent to defueling of the reactor vessel and the reactor coolant system but prior to completion of the core debris shipping program. The possibility of criticality in the Reactor Building (RB) is precluded and no canisters containing core material are in the RB.

Mode 3 - The period subsequent to shipment of the remaining core material offsite.

Thirty days prior to an anticipated change in mode, the licensee proposes to submit to the NRC a report which provides the basis for the transition.

As noted above the licensee has defined Mode 1 as the current cleanup condition and Mode 2 would begin following the completion of defueling. The licensee's Mode 1 defueling program is expected to result in the removal of greater than 99% of the reactor fuel. During Mode 1 all technical specification requirements, with one exception, currently in the license would be maintained. This exception involves the licensee's proposal to immediately delete the requirement for NRC approval prior to changes in their Radiation Protection Plan.

After the transition from Mode 1 to Mode 2 the systems and requirements for monitoring and protecting the reactor core are no longer needed and the licensee proposes their deletion. The Borated Water Injection Capability, Reactor Coolant System Water Control, Reactor Coolant System Temperature Control and Neutron Monitoring are examples of systems and monitoring capability which would no longer be needed. Additionally, the requirement for licensed operators would no longer be needed once the core material is removed. Also, the licensee proposes to delete the requirement for preapproval of procedures by the NRC.

The transition from Mode 2 to Mode 3 would further reduce the operability requirements for certain plant systems. For example, with the removal offsite of all of the defueled core material, the requirements to maintain a specific water level and boron concentration for Spent Fuel Pool 'A' would be deleted. Additionally, limitations on crane operation inside the Fuel Handling Building would be deleted and the requirement for the collection of meteorological data would be eliminated from the Technical Specifications.

The licensee also proposes a number of administrative changes to the Technical Specifications. Sections that have been deleted in previous license amendments would be completely removed from the Technical Specifications and no mention of their past incorporation in the license would appear in the amended Technical Specification. Section 2, Safety Limits, previously deleted, would be revised to state that there are no safety limits applicable to TMI-2. The licensee also proposes to revise the Technical Specification Index to be consistent with the deletions. No extensive renumbering of the remaining technical specifications is proposed.

The licensee proposes to change the applicability of Technical Specifications 3.3.3.1, Radiation Monitoring Instrumentation; 3.3.3.8, Fire Detection Instrumentation; 3.6.1.4, Internal Pressure; 3.6.3.1, Containment Purge Exhaust System; 3.7.6.1, Flood Protection; 3.7.9, Sealed Sources; 3.7.10.1, Fire Suppression Water System; 3.7.10.2, Deluge/Sprinkler Systems; 3.7.10.4, Fire Hose Stations; 3.7.11, Penetration Fire Barriers; 3.9.12.1, Fuel Handling Building Air Cleanup Exhaust System; 3.9.12.2, Auxiliary Building Air Cleanup Exhaust System and 3.9.13, Accident Generated Water from "Recovery Mode," which is the current term for the ongoing cleanup operations, to "Modes 1, 2, 3" which is the licensees' proposed terminology for the remainder of the current cleanup effort. Accordingly, there would be a change in the terminology but not in the applicability of the requirements. To parallel this change the licensee proposes to delete from section 1.3 the definition of "Recovery Mode" and replace it with the three modes discussed above.

A revised definition of "Containment Integrity" (Section 1.7) is proposed. The new definition is consistent with the current definition but has been modified to define specific criteria under which double valve isolation external to containment would be allowed due to the unique circumstances of TMI-2. These criteria are similar to those which fall under the present provision of allowing double valve isolation outside containment in accordance with NRC approval.

A new definition of "Containment Isolation" (Section 1.21) has been added to the Definitions Section. The licensee proposes to add this definition to support the addition of Technical Specification 3.6.1.2. Containment isolation requirements have been added for Modes 2 and 3 to provide provisions for maintaining the containment as a contamination barrier during these two facility modes.

The licensee proposes to change the phase "Recovery Mode" to "Facility Mode" in Section 3.0.1, Limiting Conditions for Operation, to reflect the use of modes in the applicability of certain Technical Specifications.

Proposed changes to Technical Specification 3.1.1.1, Borated Cooling Water Injection, incorporate a minimum temperature requirement in the action statement and applicability of this specification only during Mode 1. Borated cooling water injection capability to the Reactor Coolant System (RCS) to eliminate the possibility of an inadvertent criticality is only applicable if there is fuel in the RCS. Once Mode 1 defueling is completed there is no requirement for borated water injection. The minimum temperature requirement was added to the action statement to be consistent with the minimum temperature requirements elsewhere in the specification.

The licensee proposes to make Technical Specification 3.1.1.2, Boron Concentration Reactor Coolant System, applicable only during Mode 1. Boration of the RCS is for the prevention of an inadvertent criticality. Once Mode 1 defueling is completed the possibility of an inadvertent criticality is eliminated, therefore, boration of the RCS is unnecessary.

The licensee proposes to modify Technical Specification 3.1.1.3, Fuel Transfer Canal and Fuel Storage Pool A Boron Concentration, which currently specifies boration of the Fuel Transfer Canal and the Spent Fuel Storage Pool "A" by removing the requirement for boration of the Spent Fuel Pool and placing it in a new section 3.1.1.4, Boron Concentration-Spent Fuel Pool "A", and making the remainder of 3.1.1.3, pertaining only to the fuel transfer canal, and applicable only during Mode 1. Once Mode 1 defueling is completed there is no further need for the fuel transfer canal and therefore, no boration requirements are necessary to avert criticality. During Mode 2 there may still be canisters containing core debris in the Spent Fuel Pool "A". Therefore, there would be a continuing requirement to maintain boration of the fuel pool. The proposed section 3.1.1.4, pertaining only to Spent Fuel Pool "A" would be applicable during Modes 1 and 2.

The licensee proposes to make Technical Specification 3.3.1.1, Intermediate and Source Range Neutron Flux Monitors, applicable only during Mode 1. Once Mode 1 defueling is completed, the shutdown status of the core is assured and the basis for maintaining these monitors no longer exists. The licensee also proposes to delete from the action statement the requirement that a Special Report be submitted if the monitors are inoperative. According to 10 CFR 50.73 a licensee is required to submit a Licensee Event Report (LER) when a Technical Specification Action Statement has not been satisfied. The requirement for a special report is redundant with the requirements under 10 CFR 50.73. The LER would contain the same information as required by the Special Report.

Technical Specifications 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation, would be deleted by the licensee. The current technical specification requires the operability of one ESFAS channel related to the automatic starting of the diesel generators with loss of offsite power. Amendment 27 deleted the operability requirements for the diesel generators. There is no longer a basis for maintaining the ESFAS Instrumentation operable.

The licensee proposes to change the applicability of Technical Specification 3.3.3.4, Meteorological Instrumentation, from Recovery Mode to Modes 1 and 2 and the time clock in the action statement would be changed from eight hours to seven days. The potential off-site consequences of the worst case accident during Mode 3, a fire in the reactor building, is bounded by the numerical guidelines of 10 CFR 50 Appendix I. Since the basis for requiring meteorological data is to evaluate the need for initiating protective measures to protect the health and safety of the public and the worst case release is less than the releases permitted under Appendix I no protective measures would be necessary. Therefore there would be no requirement to maintain meteorological instrumentation for TMI-2. Changing the action statement requiring an inoperable meteorological monitoring channel to be restored within a specified period of time from 8 hours to 7 days is consistent with the requirements of the B&W Standard Technical Specifications and the TMI-2 pre-accident Technical Specification. The requirement for the eight hour timeclock was incorporated in Technical Specification 3.3.3.4 by the NRC Amendment of Order dated February 11, 1980. At the time of this Order, the Reactor Building contained high concentration of radioactive Krypton-85, as well as many other radionuclides. In the event of a leak from the facility it would have been and has been important to have operable meteorological instrumentation to assess the consequences of the release. As the cleanup progresses the magnitude of potentially airborne radionuclides in the facility that could be released to the environment has been substantially reduced. Therefore, the licensee concludes that the original need for the rapid restoration of meteorological data channels no longer exists.

The licensee proposed to change Technical Specification 3.3.3.5, Essential Parameter Monitoring Instrumentation. The specification currently requires the monitoring of the following essential parameters: (1) reactor building pressure, (2) reactor vessel water level, (3) incore temperature, (4) reactor building water level, (5) borated water storage tank level, (6) steam generator level, (7) spent fuel storage pool "A" water level, and (8) fuel transfer canal (deep end) water level.

The licensee proposed to make parameters 3, 4, 5 and 6 applicable only to Mode 1. The requirements for reactor building pressure (1 above) has been transferred to Section 4.6.1.4.a of the Recovery Operations Plan. The requirement for reactor vessel water level (2 above) has been transferred to technical specification 3.4.2, the requirement for spent fuel storage pool "A" water level (7 above) has been transferred to technical specification 3.9.1 and the requirement for fuel transfer canal (deep end) (8 above) has been transferred to technical specification 3.9.3. Once the reactor vessel has been defueled there is no requirement for monitoring incore temperature, reactor building water level, borated water storage tank level or the steam generator level.

Technical Specification 3.3.37, Chlorine Detection Systems would be modified by the licensee by making the specification applicable only during Mode 1. Chlorine detection is required to protect the inhabitants of the control room. An accidental chlorine release would be detected promptly and the Control Room Emergency Ventilation System would automatically isolate the control room and initiate recirculation. Manning the control room will only be required during Mode 1 (see proposed changes to Section 6.2.2 below). Once the fuel has been removed from the RCS the requirement for manning the control room with licensed operators will be deleted. Therefore, the maintenance of the Chlorine Detection System would be unnecessary.

The licensee proposes to change the applicability of Technical Specification 3.4.2, Reactor Vessel Water Level Monitoring, from Recovery Mode to Mode 1. The reactor vessel water level monitor ensures that indication is available to monitor for changes in the reactor vessel water level. This device provides warning of a leak in the RCS inventory that could result in a boron dilution event. Once defueling of the reactor vessel is completed it is no longer necessary to maintain water in the reactor vessel, consequently, the capability to monitor the water level is no longer required.

The licensee proposes to change the applicability of Technical Specification 3.4.9 Pressure/Temperature Limits from Recovery Mode to Mode 1. The current specification states that the RCS shall remain open to the reactor building atmosphere and that repressurization shall only be allowed following NRC approval. Temperature limits on the RCS are specified to prevent precipitation of the boron or boiling of the Reactor coolant. Once Mode 1 defueling is completed there would be no requirement to maintain a specific boron concentration or a potential source of heat to cause boiling. Consequently, the capability to monitor the RCS water temperature and RCS pressure would not be required.

Technical Specification 3.5.1 Control Room Communications presently require that direct communications between the control room or the communication center and personnel in the reactor building be maintained. The licensee proposed to make this requirement applicable only during Mode 1 when core alterations are being made. The current specification states that it is applicable during core alterations. Once the licensee completes Mode 1 defueling there will no longer be any core, therefore, core alterations would not be possible and this requirement would not be necessary.

The licensee proposes to change Technical Specification 3.6.1.1, Containment Integrity, the current specification is applicable during the Recovery Mode the licensee has proposed making it applicable only during Mode 1. Once Mode 1 defueling is completed double containment isolation would no longer be required since the maximum possible release of radionuclides due to the worst case accident, a fire inside containment, would be less than the 10 CFR 50 Appendix I numerical guidelines. The licensee proposes to further modify the specification by allowing modifications to containment penetrations provided that a single isolation barrier is maintained. If no isolation barriers are provided the action statement requires the cessation of any activity inside the reactor building that could result in a radiation release.

Technical Specification 3.6.1.3, Containment Air Locks (Mode 1), would be modified by the licensee to be applicable only during Mode 1. The specification requires the operability of each air lock and both air lock doors. If an air lock is inoperable the requirement is to maintain at least one door closed and repair the air lock to operable status within 24 hours. Once Mode 1 defueling is completed the source term for an inadvertent release of radioactivity to the environment is substantially reduced, consequently, the need to restore the air lock to operable status would not be required.

The licensee proposes to add Technical Specification 3.6.1.2, Containment Isolation. The proposed Technical Specification would require primary containment isolation during Modes 2 and 3. The specification would provide an appropriate provision for maintaining the containment as a contamination barrier subsequent to Mode 1 defueling. There would no longer be a requirement for double isolation of penetrations.

The licensee also proposes to add Technical Specification 3.6.1.6, Containment Air Locks (Modes 2 and 3). Each containment air lock would be considered operable with at least one air lock door operable during Modes 2 and 3. This would provide an appropriate containment barrier subsequent to defueling. There would no longer be a requirement for double isolation capability for the air locks.

Technical Specification 3.6.1.5, Air Temperature specifies the primary containment average air temperature. The licensee proposes to make this specification applicable only during Mode 1. The purpose of this specification is to insure that the life of instrumentation and equipment installed in the containment is maximized, and that the boron in the RCS will remain in solution preventing the possibility of recriticality. Once defueling is completed there will no longer be the requirements for the operability of most of the instrumentation and equipment, and boration of the RCS, consequently, the requirement to maintain containment temperature within a specified range is not required.

The licensee proposes to modify Technical Specification 3.7.7.1, Control Room Emergency Air Cleanup System by changing the applicability from Recovery Mode to Mode 1. The Control Room Emergency Air Cleanup System is required to be maintained operable to protect control room operators in the event of an accident and to maintain control room habitability in the event of chemical releases. Once Mode 1 defueling is completed there will be no requirement to man the control room (see proposed changes to Section 6.2-2), consequently maintenance of control room habitability is not required.

Technical Specification 3.7.10.3, Halon System, protects circuits and equipment required for safe shutdown and core protection in specific areas of the plant from the propagation of a fire. The licensee proposes to change the applicability of this specification from Recovery Mode to Mode 1. Once Mode 1 defueling is completed there will be no circuits or equipment necessary for the protection of the core.

The licensee proposes to change the applicability of Specification 3.8.1 A.C. Sources, 3.8.2, Onsite Power Distribution Systems, and 3.8.2.3, DC Distribution. The licensee proposes to change the applicability from Recovery Mode to Mode 1. The purpose of these specifications is to assure that the power sources and associated distribution systems are available to supply the safety related equipment required to maintain the unit in a stable condition following the March 28, 1979 accident. Once Mode 1 defueling is completed no safety related equipment will be required to maintain the unit in a safe and stable condition, consequently, the power sources and distribution systems would not be required. The licensee also proposes to administratively renumber specifications 3.8.2.1.b, and 3.8.2.3 and 3.8.2.2.1 respectively.

Technical Specifications 3.9.1, Spent Fuel Pool "A" Water Level Monitoring and 3.9.2 Spent Fuel Pool "A" Water Level require monitoring of the water level in the spent fuel pool and maintenance of a level specified in the Recovery Operations Plan. The licensee proposes to establish the applicability to these two specifications to Modes 1 and 2. Spent Fuel Storage Pool "A" is used to store defueling canisters containing core material prior to shipment offsite. While canisters are in storage in the spent fuel pool the fuel pool will be flooded and borated. Once all canisters have been removed from the TMI site the spent fuel pool will no longer be used, consequently, monitoring and maintenance of a specific water level would no longer be required.

The licensee proposes to change Technical Specifications 3.9.3, Fuel Transfer Canal (Deep End) Water Level Monitoring, and 3.9.4, Fuel Transfer Canal (Deep End) Water Level, by establishing the applicability of the specifications to Mode 1. The Fuel Transfer Canal is used to transfer defueling canisters from the reactor building to the fuel handling building. While transfers take place the Fuel Transfer Canal is flooded and borated to protect personnel from radiation. Once Mode 1 defueling is completed the Fuel Transfer Canal will no longer be needed to transfer defueling canisters and, therefore, monitoring and maintenance of a specific water level would no longer be required.

Technical Specification 3.10.1, Crane Operations-Containment Building, delimits load travel within containment. The license proposes to change the applicability of the specification from Recovery Mode to Mode 1. The basis for this specification is to prevent a load drop into the reactor vessel causing a reconfiguration of the core debris and/or structural damage which could hinder defueling. Once Mode 1 defueling is completed the basis for controlling heavy loads inside the containment will be eliminated and the specification will no longer be required.

The licensee proposes to change the applicability of Technical Specification 3.10.2, Crane Operations-Fuel Handling Building, from Recovery Mode to Modes 1 and 2. The basis for this specification is to prevent a load drop in the Fuel Handling Building causing damage to canisters containing core material. Subsequent to Modes 1 and 2 all core material will have been shipped off-site. Thus, the basis for controlling heavy loads inside the Fuel Handling Building is eliminated and the specification is not required.

The licensee proposes to clarify the applicability of portions of Technical Specification 6.2-2, TMI-2 Organization. Specification 6.2.2 specifies, in part, the staffing required by 10 CFR 50.54 paragraphs (m)(2) (ii) and (m)(2)(iii) for fueled nuclear power plants. Subsequent to Mode 1 defueling TMI-2 will no longer be considered fueled and the licensee proposes that the requirements for licensed operators will no longer apply. Furthermore, the current specification states that a licensed operator would be in the control room when there is fuel in the reactor. The licensee proposes that specifying the applicability to only Mode 1, when there is fuel in the reactor, is an administrative change consistent with NRC regulations. Specification 6.2.2.C states that an individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The licensee proposes to change the wording of this specification to "During Mode 1, an individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor."

The licensee proposes to limit the applicability of Technical Specification 6.8.2.2, Procedures, to Mode 1. The current Technical Specification requires NRC review and approval of all procedures and changes thereto which alter the distribution or processing of a quantity of radioactive material the release of which could cause the magnitude of radiological releases to exceed 10 CFR 50 Appendix I limits. Once Mode 1 defueling is completed the potential source term within the reactor building and the maximum credible accident, a fire, would not result in a maximum dose to an individual from fission products and transuranics in excess of 10 CFR Appendix I limits. Although during Mode 2 there will be defueling canisters containing significant amounts of fuel still onsite the licensee is of the opinion that these canisters have proven to be effective in preventing inadvertent criticality.

Furthermore, potential accident scenarios associated with these canisters have shown that they provide adequate protection for the public.

The licensee proposes to change Technical Specification 6.11, Radiation Protection Program, by deleting the requirement for NRC approval of the Radiation Protection Plan. Removal of this requirement is based on the past performance of the licensee in the area of radiation protection and the desire to remove the NRC from the procedure review and approval cycle at TMI-2. It is also consistent with the Standard Technical Specifications for Babcock and Wilcox plants. Auditing by the NRC of the Radiation Protection Plan and licensee compliance with the plan would continue.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if 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, (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.

TMI-2 is in a long-term cold shutdown for accident recovery. Short-lived fission products which make up the preponderance of the source term for operating reactors have decayed to negligible levels. The decay heat produced by the core has now dropped to less than 10 kilowatts and forced cooling of the core has not been required or used since 1981. Consequently, in previous license amendments, the staff has determined that the potential accidents analyzed for TMI-2 in the current mode are bounded in scope and severity by the range of accidents originally analyzed in the facility FSAR.

The changes proposed by the licensee are changes to the Appendix A Technical Specifications. They consist primarily of specifying the circumstances under which the existing Specifications are applicable and improving the clarity of the requirements. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated because no changes to current safety systems or setpoints are proposed while there is still sufficient fuel in the RCS to cause a criticality event. No active systems are required to maintain TMI-2 in its current safe shutdown condition. Once Mode 1 defueling is completed the possibility of an offsite release of radiation in excess of 10 CFR 50 Appendix I limits is greatly reduced. Those systems necessary to monitor the core and facilitate defueling will no longer be required. Maintenance of these systems will no longer be necessary.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because no new modes of operation or new equipment are being introduced. Deletion of monitoring requirements cannot create the possibility of any new or different kind of accident. Deletion of safety systems designed to protect the core once the core is removed cannot increase the probability of accidents. The proposed changes represent a gradual reduction in the scope of license requirements and are consistent with the changing status of the facility as the cleanup progresses.

The proposed changes do not involve a significant reduction in a margin of safety, because, as mentioned perviously, no active components are required to maintain the current safe shutdown of TMI-2. Furthermore, as the cleanup progresses the margin of safety increases. Once Mode 1 defueling and Mode 2 offsite shipment is completed there will be a significant increase in the margin of safety.

Based on the above considerations, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: State Library of Pennsylvania Government Publications Section, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

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

NRC Project Director: William D. Travers

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia

Date of amendment request: July 13, 1987

Description of amendment request: This amendment would modify a provision of Section 1.II of the Hatch Unit 1 Technical Specifications (TS) which limits the operating cycle length for instrument and electrical surveillance. The existing TS 1.II limits the operating cycle interval to 15 months as regards instrument and electrical surveillance. The proposed change would define the operating cycle interval as 18 months instead of 15 months and would delete the reference to electrical and instrumentation surveillance.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR Part 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if 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 new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Standard Technical Specifications (STS) include a Surveillance Frequency Notations table (Table 1.1) which defines a refueling cycle interval as 18 months. A copy of this table is incorporated in the TS for Hatch Unit 2. However, the Hatch Unit 1 TS, which are in an earlier "custom" TS, have a Section 1.II defining "Surveillance Frequency" in which it is stated that: "The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months." The terms operating cycle and refueling cycle are synonymous. Thus, the 15-month operating cycle for Unit 1, as specified in TS 1.II, is more restrictive than the 18-month refueling cycles specified in the STS and in the Hatch Unit 2 TS. Further complicating matters, Amendment 110 to the Hatch Unit 1 TS added a Table 1.1, "Frequency Notations," in which a refueling cycle interval is specified as 18-months. However, Section 1.II was not changed. Thus, the Unit 1 TS are internally inconsistent. The licensee proposes to change the words in Hatch Unit 1 definition of Surveillance Frequency to: "The operating interval is defined as 18-months." This change would remove the internal inconsistency and would adjust the operating cycle for Hatch Unit 1 to the same 18-month period allowed for Unit 2.

The licensee states that the actual plant trip setpoints for instruments and electrical equipment are set conservative to the TS allowable values, such that the allowable values are not compromised during an operating cycle by instrument drift. Extending the allowable time between refuelings to 18 months instead of 15 months would require an adjustment to the actual trip setpoints, but would not affect the TS allowable setpoints. Thus, this change would not involve a significant increase in the probability or consequences of an accident previously evaluated. Further, since the design functions of the electrical and instrument systems are not affected by this change, the change would not create the possibility of a new or different kind of accident from any accident previously evaluated. Finally, margins of safety are not significantly reduced by the proposed change since the TS allowable setpoints are unchanged.

On the basis of the above, the Commission has determined that the requested amendment meets the three criteria and, therefore, has made a proposed determination that the amendment application does not involve a significant hazards consideration.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

Attorney for licensee: Bruce W. Churchill, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Lawrence P. Crocker, Acting Project Director

GPU Nuclear Corporation, Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: November 9, 1987 (TSCR 173)

Description of amendment request: The proposed amendment would remove the Radiological Environmental Monitoring Program (REMP) from the Technical Specifications. In addition, the REMP would be changed to: (a) permit monitoring frequencies to be changed to calendar periods such as weekly, monthly, etc. (b) increase the time from 30 days to 60 days for reporting environmental samples exceeding the reporting levels; and (c) make typographical and administrative changes. The REMP will continue to be required by the Technical Specifications even though it is not in the Technical Specifications. Future changes to the REMP that would reduce the effectiveness of the REMP are required to be reviewed and approved by NRC prior to implementation by the licensee.

Basis for proposed no significant hazards consideration determination: The licensee proposed Technical Specification Change Request (TSCR) No. 173 to remove the REMP from the Technical Specifications although the REMP will still be required by the Technical Specifications. It has evaluated TSCR 173 to determine if a significant hazards consideration exists. The results of this evaluation are given below in terms of the criteria in 10 CFR 50.92(c):

Removal of the Radiological Environmental Monitoring Program (REMP) from the Technical Specifications reduces the size of the Technical Specifications without impacting the effectiveness of the REMP. The changes to the REMP are administrative with one noted change to the reporting requirements of the REMP. The REMP will continue to be required by technical specification even though 10 CFR 50.36a does not require the REMP to be in the technical specifications. The REMP is a reformatted version of the technical specification it replaces. The information and specific requirements of the program are essentially the same as the former technical specification with the following exceptions:

1. Monitoring frequencies that are specified in a specific number of days has been changed to a calendar period such as weekly, monthly, etc. as appropriate. Other minor changes have been made to be more consistent with NUREG-0472 and the Branch Technical Position.

2. Typographical errors have been corrected.

3. A reporting requirement for environmental samples exceeding the reporting levels as specified in Table 2 has changed. This was changed from 30 to 60 days to allow adequate time for laboratory analysis of samples.

Safety and safety controls shall remain unaffected.

Future changes that reduce the effectiveness of this initially approved REMP shall be reviewed and approved by the NRC prior to implementation by GPUN. This is required by Technical Specification 6.15.

Future changes that do not reduce the effectiveness of the REMP shall be submitted to the NRC for review in the Annual Radiological Environmental Operating Report for the period in which the changes were made. These changes will be fully reviewed and approved by GPUN management consistent with review and approval procedures, prior to implementation. This is required by Technical Specification 6.5.

GPUN has determined that this technical specification change request poses no significant hazards as defined by the NRC in 10 CFR 50.92.

Since this change is administrative:

A. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The technical specification changes are administrative and do not affect plant equipment. The results of this change will not impact the safety of the plant or the public health.

Therefore, the technical specification change for the Radiological Environmental Monitoring Program does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

B. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of affect from any accident previously evaluated since it does not affect plant equipment.

Therefore, it is concluded that the technical specification change for the Radiological Environmental Monitoring Program does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. All safety criteria as described in the former technical specification bases are preserved in the Radiological Environmental Monitoring Program.

Therefore, it is concluded that the technical specification change for the Radiological Environmental Monitoring Program does not involve a significant reduction in a margin of safety.

We agree with GPU's conclusion that this license amendment request involves no significant hazards considerations in that operation of TMI-1 in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of any 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 REMP will remain as a functional program and we can perceive at this time no significant hazard from removing the REMP from the Technical Specifications. Adjustments to monitoring frequencies and one reporting requirement are minor and insignificant in terms of plant safety and public health. Future changes to the REMP that would reduce its effectiveness are required to be approved by NRC prior to implementation.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Therefore, the staff proposes to determine that the application for amendment involves no significant hazards consideration.

Local Public Document Room location: Government Publications

Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126

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

NRC Project Director: John F. Stolz

Louisiana Power and Light Company, Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: December 10, 1987.

Description of amendment request: The proposed change would revise Technical Specification 3.5.2, ECCS Subsystems - Tavg Greater than 350° F and Technical Specification 3.5.3, ECCS Subsystems - Tavg Less than 350° F by adding a note to the Applicability section of both Technical Specifications to indicate that two ECCS subsystems are required to be operable when the RCS average temperature is equal to or greater than 500° F.

Technical Specification 3.5.2 currently requires two independent emergency core cooling system (ECCS) subsystems to be operable when the reactor is in Modes 1, 2 and 3; however, the requirements of this Technical Specification in Mode 3 are applicable only if the pressurizer pressure is equal to or greater than 1750 psia. The proposed change will add a note to the Mode 3 applicability statement that will require both ECCS subsystems to be operable any time the RCS average temperature is equal to or greater than 500° F, regardless of the pressurizer pressure.

Technical Specification 3.5.3 currently requires one ECCS subsystem to be operable if the reactor is in Modes 3 and 4 with a Mode 3 requirement that the pressurizer pressure is less than 1750 psia. The proposed change to this Technical Specification is similar to the proposed change to Technical Specification 3.5.2 in that a note will be added to the Mode 3 applicability statement that requires the RCS average temperature to be less than 500° F before it is acceptable to have only one ECCS subsystem in service.

The reason for the proposed change to these Technical Specifications is to ensure that at least one train of high pressure safety injection (HPSI) is available (even if a single failure is assumed) to mitigate the consequences of a postulated steam line break (SLB) accident initiated from an RCS average temperature of 500° F or greater. The Cycle 2 safety analysis has shown that borated water from HPSI is required to prevent the core from becoming critical during the uncontrolled RCS cooldown (associated with a SLB) from greater than 500° F.

In addition, the proposed change will also revise the title of the subject Technical Specifications such that they will be described in terms of modes of operation rather than average coolant temperature.

Basis for Proposed No Significant Hazards Considerations Determination: The NRC staff proposes that the proposed change does not involve a significant hazards consideration because, as required by the criteria of 10 CFR 50.92(c), operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of any 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 the margin of safety. The basis for this proposed finding is given below.

(1) The proposed change will require that two ECCS subsystems are operable whenever the average temperature of the RCS is equal to or greater than 500° F. This will ensure that, even if one ECCS subsystem is assumed to fail, one train of HPSI will be available to inject borated water into the RCS during an SLB. As described in the safety analysis for Cycle 2, borated water (from HPSI) is required to mitigate the reactivity transient associated with the RCS cooldown and prevent the core from returing to a critical condition. Below 500° F the RCS cooldown (and associated reactivity transient) during the SLB is less severe and HPSI flow is not required to maintain the core subcritical. Therefore, since the proposed change reduces the consequences of a SLB it will not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed change does not involve any physical changes to plant systems, structures or components nor will there be any significant changes to plant operating procedures. The proposed change will simply clarify the RCS conditions which must exist prior to taking one of the ECCS subsystems out of service. Thus, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The intent of this Specification is to ensure there will be sufficient emergency core cooling capability available in the event of a LOCA and a coincident single failure that results in the complete loss of one ECCS subsystem. The proposed change will not affect the LOCA analysis since it merely adds a restriction that requires both ECCS subsystems to be operable whenever the RCS temperature is equal to or greater than 500° F. This additional restriction ensures that sufficient borated water can be added to the RCS to mitigate the reactivity transient associated with the uncontrolled RCS cooldown that occurs during a steam line break. Since the proposed change adds a restriction that was not already a part of the Technical Specifications and since this restriction ensures that the consequences of a broader range of steam line breaks can be mitigated, the proposed change will result in an increase in the margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether significant hazards consideration exists by providing certain examples (52 FR 7751) of amendments that are considered not likely to involve significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications (e.g., a more stringent surveillance requirement).

In this case, the proposed change is similar to Example (ii) in that it constitutes an additional restriction (i.e., RCS temperature) that must be satisfied before it is acceptable to have only one ECCS subsystem in service.

The staff has reviewed the licensee's no significant hazards consideration analysis. Based on the review and above discussions, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room Location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122

Attorney for licensee: Bruce W. Churchill, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N St., NW., Washington, DC 20037

NRC Project Director: Jose A. Calvo

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit No. 2, Scriba, New York

Date of amendment request: November 16, 1987

Description of amendment request: The proposed amendment would revise the allowable value and isolation trip setpoints for the reactor core isolation cooling (RCIC) high steam line flow. As noted in the Technical Specifications, the existing values are preliminary with the actual values to be determined during the startup test program. The proposed changes are based on system testing during the startup test program. The proposed amendment is in accordance with the licensee's application of November 16, 1987.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a 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 proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated for the following reasons:

The RCIC Steam Line break analysis assumes that the system will isolate when the steam flow reaches 300% of rated steam flow. This change to the Technical Specification assures that the as-built plant is in agreement with the design basis. Revising the setpoint to the as-built conditions equivalent to the 300% rated flow value assures that a RCIC steam line break will be detected and isolated in accordance with the requirements of GDC 54 without impacting the qualification or operation of other safety systems or safe shutdown of the plant. The new setpoint is conservative relative to the old setpoint. In summary, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes will not create the possibility of a new or different kind of accident previously evaluated for the following reasons:

The reactor building response to previously evaluated accidents remains within previously assessed limits of temperature and pressure. Further, all safety-related systems and components remain within their applicable design limits. Thus, system and component performance is not adversely affected by this change, thereby assuring that the design capabilities of those systems and components are not challenged in a manner not previously assessed so as to create the possibility of a new or different kind of accident.

In addition, since the design basis for RCIC system isolation has not changed, the environmental qualification of plant equipment is not adversely affected by this proposed amendment, further assuring that components are not challenged in a manner not previously assessed. In summary, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes will not involve a significant reduction in a margin of safety for the following reasons:

The proposed change will not cause existing Technical Specification operational limits or system performance criteria to be exceeded. The proposed change ensures that the system design requirements are met. Allowances for instrument drift, instrument accuracy, and calibration capability have been maintained in accordance with Bases Section B3/4.3.2 of the Technical Specifications. Therefore, the proposed change does not result in a significant reduction in a margin of safety.

Based upon the above considerations, the staff proposes to determine that the proposed changes do not constitute a significant hazards consideration.

Local Public Document Room location: Penfield Library, State University College, Oswego, New York 13126.

Attorney for licensee: Mr. Mark Wetterhahn, Esq., Conner & Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, DC 20006.

NRC Project Director: Robert A. Capra, Director

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

Date of amendment request: December 4, 1987

Description of amendment request: The amendment would revise Technical Specification Section 3/4.3.2 to delete the chlorine detection system. The chlorination systems at Millstone Unit Nos. 1, 2 and 3 have been modified to use sodium hypochlorite instead of gaseous chlorine. This has resulted in the elimination of on-site bulk storage of liquid chlorine and the possibility of an on-site chlorine release.

Basis for proposed no significant hazards consideration determination: In accordance with 10 CFR 50.92, the licensee has reviewed the proposed changes and has concluded that the amendment does not involve a significant hazards consideration because the change would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed. The potential for a chlorine release affecting control room habitability no longer exists since the chlorine rail cars have been removed from the Millstone site. Thus, removal of the requirements on the chlorine detection system will not increase the consequences of any event.

2. Create the possibility of a new or different kind of accident from any previously analyzed. There are no changes in the way the plant is operated. No new failure modes are introduced.

3. Involve a significant reduction in a margin of safety. Control room habitability is not affected because on-site chlorine bulk storage has been eliminated, the number of chlorine rail, truck, and barge shipments does not exceed the levels discussed in Regulatory Guide 1.78, and the credible off-site chlorine bulk storage facilities are at least 5 miles distant from the site. The proposed changes do not affect the consequences of any accident previously analyzed.

The staff has reviewed the licensee's submittal and concurs with its no significant hazards determination.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry and Howard, One Constitution Plaza, Hartford, Connecticut 06103-3499.

NRC Project Director: John F. Stolz

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

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 and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with these actions was published in the **Federal Register** as indicated. No request for a hearing or petition for leave to intervene was filed following this notice.

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 amendments, (2) the amendments, and (3) the Commission's related letters, Safety Evaluations and/or Environmental Assessments as indicated. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects.

Carolina Power & Light Company, Dockets Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: November 25, 1985 and supplemented October 15, 1987

Brief description of amendments: The amendment relocates a footnote from item 1.c.1 of Table 3.3.2-1 to item 1.c.1 of Table 4.3.2-1, thereby ensuring that the required surveillance testing of mechanical pumps is identified.

Date of issuance: December 30, 1987

Effective date: December 30, 1987

Amendments Nos.: 115 and 142

Facility Operating License Nos. DPR-71 and DPR-62. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1986 (51 FR 3710) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1987.

No significant hazards consideration comments received: No.

The October 15, 1987 letter provided corrected technical specification pages that did not change the initial determination of no significant hazards consideration as published in the **Federal Register.**

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: January 28, 1987

Description of amendments: These amendments revise the Technical Specifications to incorporate additional action steps describing steps operators should take if core flow and power do not meet the definition of recirculation system operability and to change the surveillance requirements to require that baseline average power range monitor and local power monitor neutron flux noise levels be established after each refueling outage.

Date of issuance: December 30, 1987

Effective date: December 30, 1987

Amendments Nos.: 114 and 141

Facility Operating License Nos. DPR-71 and DPR-62. Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** June 17, 1987 (52 FR 23097) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1987.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

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

Date of application for amendments: October 8, 1987

Brief description of amendments: The amendments modify the Technical Specifications defining fuel Average Planar Linear Heat Generation Rate limits and Emergency Core Cooling System surveillance requirements.

Date of issuance: December 21, 1987

Effective date: December 21, 1987

Amendment Nos.: 150 and 87

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

Date of initial notice in **Federal Register:** November 18, 1987 (52 FR 44244) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 21, 1983.

No significant hazards consideration comments received: No

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

Date of application for amendments: January 9, 1987

Brief description of amendments: The amendments revised the Technical Specifications by deleting the provision that the auxiliary building crane main hoist be deenergized and the load blocks unloaded whenever the crane is moved over the spent fuel assemblies in the spent fuel pool.

Date of issuance: December 17, 1987.

Effective date: December 17, 1987

Amendment Nos.: 113 and 96.

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 28, 1987 (52 FR 2883) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 1987.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maude Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Indiana and Michigan Power Company, Docket No. 50-316, Donald C. Cook Nuclear Plant, Unit No. 2, Berrien County, Michigan

Date of application for amendment: October 28, 1987

Brief description of amendment: The amendment revised the provisions in the Technical Specifications to extend 18-month surveillances from December 31, 1987 to the refueling outage currently scheduled to begin in early 1988 for response-time testing for reactor trip and engineering safety features (ESF) instrumentation; response testing of equipment to ESF signals; reactor vessel level indication calibration; auxiliary feedwater system testing, including channel functional testing of loss of main feedwater pump signal; and diesel generator testing, including relief valve testing and essential service water valve testing.

Date of issuance: December 28, 1987

Effective date: December 28, 1987

Amendment No.: 97

Facility Operating License No. DPR-74. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 27, 1987 (52 FR 45413) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 28, 1987.

No significant hazards consideration comments received: No. The proposed amendment was noticed with an opportunity for prior hearing.

Local Public Document Room location: Maude Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Mississippi Power & Light Company, System Energy Resources, Inc., South Mississippi Electric Power Association, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Dates of applications for amendment: October 17, 1986 and August 6, 1987, as supplemented December 15, 1987.

Brief description of amendment: The August 6, 1987 application for license amendment requested changes to the Technical Specifications (TS), Appendix A to the operating license, in eight areas: (1) a clarification to the definition of secondary containment integrity; (2) a change in the name of a supporting organization represented on the Safety Review Committee; (3) a nomenclature change for a secondary containment isolation valve; (4) deletion of the manual initiation handswitch calibration requirement for ECCS pumps; (5) deletion of expired footnotes; (6) a change to reflect new upper containment pool gates; (7) a change to add certain smoke detectors; and (8) a modification to the setpoint for residual heat removal (RHR)/reactor core isolation cooling (RCIC) steam line high flow. These changes are made in this amendment. The October 17, 1986 application for license amendment requested four changes to the TS. Three of the changes were made in Amendment 29 to the operating license, issued March 31, 1987. The fourth requested change, the addition of TS for smoke detectors in the control rod drive repair room, is made in this amendment.

Date of issuance: December 30, 1987

Effective date: December 30, 1987

Amendment No. 42

Facility Operating License No. NPF-29. This amendment revises the Technical Specifications and the Environmental Protection Plan.

Dates of initial notice in **Federal Register**: September 23, 1987 (52 FR 35796) The December 15, 1987 letter provided supplemental information which did not change the initial determination of no significant hazards considerations as published in the **Federal Register**. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1987.

No significant hazards consideration comments received: No

Local Public Document Room location: Hinds Junior College, McLendon Library, Raymond, Mississippi 39154

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

Date of amendment request: October 20, 1987

Brief description of amendment: The amendment changed the Technical Specifications relating to design features of the fuel storage facilities.

Date of issuance: December 21, 1987.

Effective date: December 21, 1987.

Amendment No.: 113

Facility Operating License No. DPR-46. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1987 (52 FR 44246). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1987.

No significant hazards consideration comments received: No.

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305.

Northeast Nuclear Energy Company, et al., Docket No. 50-245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of application for amendment: October 20, 1987

Brief description of amendment: To reflect deletion of low reactor pressure permissive switches from the emergency core cooling system (core spray and low pressure coolant injection) pump start logic.

Date of issuance: December 17, 1987

Effective date: December 17, 1987

Amendment No.: 13

Facility Operating License No. DPR-21. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 13, 1987 (52 FR 43694). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 17, 1987

No significant hazards consideration comments received: No.

Local Public Document Room location: Waterford Public Library, Rope Ferry Road, Waterford, Connecticut.

Vermont Yankee Nuclear Power Corporation, Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: January 16, 1987

Brief Description of amendment: The amendment changes Technical Specifications to clarify and enhance limiting conditions of operation and surveillance requirements pertaining to the standby liquid control system.

Date of issuance: December 30, 1987

Effective date: 30 days from date of issuance

Amendment No.: 102

Facility Operating License No. DPR-28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 12, 1987 (52 FR 7700) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1987.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301.

Vermont Yankee Nuclear Power Corporation, Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: April 28, 1987 as clarified by letter dated November 2, 1987.

Brief Description of amendment: This amendment revises the Technical Specifications to reflect administrative changes to Section 6 of the Technical Specifications.

Date of issuance: December 29, 1987

Effective date: December 29, 1987

Amendment No.: 101

Facility Operating License No. DPR-28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 23, 1987 (52 FR 35808) and renoticed on November 18, 1987 (52 FR 44247). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 29, 1987

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION AND OPPORTUNITY FOR HEARING (EXIGENT 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 and Proposed No Significant Hazards Consideration Determination and Opportunity for 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 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, 1717 H Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendments. By February 12, 1988, 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 petition for leave to intervene. Requests for a hearing and petitions 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. 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 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 fifteen (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 fifteen (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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

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

Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW., Washington, DC, by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to (*Project Director*): petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel-Bethesda, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for 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).

Mississippi Power & Light Company, System Energy Resources, Inc., South Mississippi Electric Power Association, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Dates of application for amendment: August 13, 1987, as revised October 23, November 25, December 22, and December 27, 1987

Brief description of amendment: The amendment provides interim changes to the Technical Specifications for the standby liquid control system and the ATWS recirculation pump trip system to reflect modifications made to conform to 10 CFR 50.62 regarding anticipated transients without scram (ATWS).

Date of issuance: December 30, 1987

Effective date: December 30, 1987

Amendment No. 41

Facility Operating License No. NPF-29: This amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** December 4, 1987 (52 FR 46136) The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated December 30, 1987.

No significant hazards consideration comment received: No

Local Public Document Room location: Hinds Junior College, McLendon Library, Raymond, Mississippi 39154.

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

Date of amendment request: December 21, 1987.

Brief description of amendment: The amendment changed the Technical Specifications to extend the secondary containment isolation logic functional test interval from six months to eighteen months.

Date of issuance: December 22, 1987.

Effective date: December 22, 1987.

Amendment No.: 114

Facility Operating License No. DPR-46. Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, consultation with State of Nebraska, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated December 22, 1987.

Attorney for licensee: Mr. G. D. Watson, Nebraska Public Power District, Post Office Box 499, Columbus, Nebraska 68601

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305.

NRC Project Director: Jose A. Calvo

Dated at Bethesda, Maryland this 7th day of January 1988.

For the Nuclear Regulatory Commission

Steven A. Varga,

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

[Doc. 88-523 Filed 1-12-88; 8:45 am]

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