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U.S. Nuclear Regulatory Commission
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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
REACTOR PROTECTION SYSTEM INSTRUMENTATION, CONTROL ROD BLOCK INSTRUMENTATION, SOURCE RANGE MONITORS AND POWER DISTRIBUTION LIMITS SURVEILLANCE REQUIREMENTS
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications (TS) for the Hope Creek Generating Station. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The changes to Surveillance Requirements proposed in this license change request will minimize unnecessary testing while continuing to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met. The proposed changes are consistent with NUREG-1433, "Standard Technical Specifications (STS) General Electric Plants, BWR/4," Revision 2. The NRC has approved similar changes for other licensees as described in Attachment 1.

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1 to this letter. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 2.

A001

PSEG requests approval of the proposed License Amendment by April 26, 2004 to be implemented within 60 days.

Should you have any questions or require additional information, please contact Mr. Paul Duke at (856) 339-1466.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 10-24-03
(date)

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Sr. Vice President - Site Operations

Attachments (2)

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS REACTOR PROTECTION SYSTEM INSTRUMENTATION, CONTROL ROD BLOCK INSTRUMENTATION, SOURCE RANGE MONITORS AND POWER DISTRIBUTION LIMITS SURVEILLANCE REQUIREMENTS

Table of Contents

1. DESCRIPTION

This letter is a request to amend Operating License NPF-57 for the Hope Creek Generating Station. The proposed amendment would revise the Technical Specification (TS) Surveillance Requirements (SRs) for certain Reactor Protection System and Control Rod Block Instrumentation, the source range monitors and power distribution limits.

The proposed changes will minimize unnecessary testing while continuing to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met. The proposed changes are consistent with NUREG-1433, "Standard Technical Specifications (STS) General Electric Plants, BWR/4," Revision 2 (Reference 1).

PSEG requests approval of the proposed License Amendment by April 26, 2004 to be implemented within 60 days.

2. PROPOSED CHANGE

The marked up pages for the proposed changes to the Technical Specifications are included in Attachment 2 of this submittal.

A. Reactor Protection System Instrumentation Surveillance Requirements
TS Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," would be revised as follows:

1. The requirements to perform the following Surveillance Requirements prior to each reactor startup would be deleted:

   a. Average Power Range Monitor (APRM) Flow Biased Simulated Thermal Power - Upscale (Functional Unit 2.b) channel functional test

   b. APRM Fixed Neutron Flux - Upscale (Functional Unit 2.c) channel functional test

   c. Intermediate Range Monitor Neutron Flux - High (Functional Unit 1.a) channel check and channel functional test

   d. Average Power Range Monitor Neutron Flux - Upscale, Setdown (Functional Unit 2.a) channel check and channel functional test

   The associated footnote (c) would also be deleted.

2. The SR for the APRM Neutron Flux - Upscale, Setdown channel functional test would be modified to allow performance to be deferred for up to 12 hours after entry into Operational Condition 2 from Operational Condition 1. The associated TS Bases would be revised to reflect the change.

B. <u>Control Rod Block Instrumentation Surveillance Requirements</u>
Technical Specification Table 4.3.6-1, "Control Rod Block Instrumentation Surveillance Requirements," would be revised as follows:

1. The requirement to perform the Rod Block Monitor (RBM) channel functional test during startup, prior to exceeding 30% of rated thermal power (RTP), if not performed within the previous 7 days would be deleted. The associated footnote (d) would also be deleted.

2. The requirements to perform channel functional tests prior to each reactor startup would be deleted for the following control rod block instrumentation trip functions:

APRM
- a. Flow Biased Neutron Flux - Upscale
- b. Inoperative
- c. Downscale
- d. Neutron Flux - Upscale, Startup

Source Range Monitors
- a. Detector not full in
- b. Upscale
- c. Inoperative
- d. Downscale

Intermediate Range Monitors
- a. Detector not full in
- b. Upscale
- c. Inoperative
- d. Downscale

Reactor Coolant System Recirculation Flow
- a. Upscale
- b. Inoperative
- c. Comparator

The associated footnote (b) would also be deleted.

3. The SR for the Reactor Mode Switch Shutdown Position functional test would be modified to allow performance to be deferred until one hour after the reactor mode switch is in the shutdown position. The associated TS Bases would be revised to reflect the change.

C. Source Range Monitor Surveillance Requirements

Source range monitor Surveillance Requirements would be revised as follows:

1. SR 4.3.7.6.b would be revised to delete the requirement to perform the source range monitor channel functional test within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days.

2. SR 4.9.2.b would be revised to delete the requirement to perform the source range monitor channel functional test within 24 hours prior to the start of core alterations.

D. Power Distribution Limits Surveillance Requirements

Surveillance Requirements for power distribution limits would be revised as follows:

1. SR 4.2.1 would be revised to require verification that all average planar linear heat generation rates (APLHGRs) are less than or equal to the limits specified in the core operating limits report (COLR) once within 12 hours after core thermal power is greater than or equal to 25% of rated thermal power (RTP) and at least once every 24 hours thereafter. The requirement to verify APLHGRs within 12 hours after completion of a thermal power increase of at least 15% of RTP would be deleted.

2. SR 4.2.3 would be revised to require verification that the minimum critical power ratio (MCPR) is greater than or equal to the applicable limit specified in the COLR once within 12 hours after core thermal power is greater than or equal to 25% of rated thermal power (RTP) and at least once every 24 hours thereafter. The requirement to verify MCPR within 12 hours after completion of a thermal power increase of at least 15% of RTP would be deleted.

3. SR 4.2.4 would be revised to require verification that the linear heat generation rates (LHGRs) are less than or equal to the applicable limit specified in the COLR once within 12 hours after core thermal power is greater than or equal to 25% of rated thermal power (RTP) and at least once every 24 hours thereafter. The requirement to verify LHGRs within 12 hours after completion of a thermal power increase of at least 15% of RTP would be deleted.

The proposed changes are consistent with NUREG-1433 and with changes previously approved by the NRC for other licensees as described in Section 4.

3. BACKGROUND

The Neutron Monitoring System (NMS) includes the SRM, IRM, APRM and Rod Block Monitor (RBM) subsystems. The SRM, IRM and APRM subsystems monitor local and core average neutron flux and trip the Reactor Protection System (RPS) when predetermined limits are reached. The SRM, IRM and APRM subsystems also generate trip signals for use in the Reactor Manual Control System (RMCS) to inhibit control rod movement (rod block) under certain conditions. The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation.

The APRM subsystem monitors neutron flux from approximately 1 percent to above 100 percent power. There are six APRM channels, each receiving core flux level signals from 21 or 22 local power range monitor (LPRM) detectors. Each APRM channel averages the 21 or 22 separate neutron flux signals from the LPRMs assigned to it, and generates a signal representing core average power. The APRM scram units are set for a reactor scram at 15 percent core power in Refuel and Startup modes.

The IRM subsystem monitors neutron flux from the upper portion of the source range to the lower portion of the power range. There are eight IRM detectors providing flux level signals to eight channels of instrumentation. The IRM neutron flux signal is applied to trip units where IRM downscale, inoperative, upscale alarm, and upscale reactor trips are generated for use in the RPS or RMCS. The detectors are normally fully inserted during startup and are withdrawn after the reactor mode selector switch is placed in "run." The mode switch is placed in "run" when the APRMs are on scale (4 to 12 percent power), ensuring IRM/APRM overlap and continuity of neutron flux monitoring.

The SRMs provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. When the IRMs are on scale, adequate information is available without the SRMs and they can be retracted.

The rod block monitor (RBM) is designed to prohibit erroneous withdrawal of a control rod during operation at high power levels. This prevents local fuel damage during a single rod withdrawal error. Because local fuel damage poses no significant threat relative to radioactive release from the plant, the RBM is a power generation system and is not used for accident mitigation.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This function prevents inadvertent criticality as the result of a control rod withdrawal during Operational Condition 3 or 4, or during Operational Condition 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

Current Technical Specification Requirements and Need for Proposed Changes

A. Reactor Protection System Instrumentation Surveillance Requirements

1. The APRM Flow Biased Simulated Thermal Power - Upscale channel functional test and the APRM Fixed Neutron Flux - Upscale channel functional test are required to be performed quarterly in Operational Condition 1. The tests are also currently required to be performed within 24 hours prior to startup, if not performed within the previous 7 days. Performing a reactor startup does not affect the ability of the APRMs to perform their function. The quarterly test frequency provides sufficient assurance the APRMs are functioning properly. Therefore, the requirement to perform the functional tests prior to startup causes the diversion of plant personnel and resources for unnecessary testing without a commensurate increase in plant safety.

   The IRM Neutron Flux - High channel functional test and the APRM Neutron Flux - Upscale, Setdown channel functional test are required to be performed weekly in Operational Condition 2. The tests are also currently required to be performed within 24 hours prior to startup, if not performed within the previous 7 days. The additional requirement is redundant to the weekly requirement since SR 4.0.4 prohibits entry into Operational Condition 2 unless the channel functional tests have been performed within the applicable surveillance interval (7 days).

   The IRM and APRM channel checks are currently required to be performed at least once per 12 hours in the applicable Operational Conditions. The checks are also currently required to be performed prior to each reactor startup. The additional requirement is redundant to the once per 12 hour requirement since SR 4.0.4 prohibits entry into the applicable Operational Conditions unless the channel checks have been performed within the applicable surveillance interval (12 hours).

2. In Operational Condition 1, the APRM Setdown channel functional test cannot be completed without using jumpers or lifted leads.

Since SR 4.0.4 prohibits the planned entry into an Operational Condition or other specified applicable condition unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified, a planned entry into Operational Condition 2 from Operational Condition 1 would be prohibited by TS if the functional test had not been performed within the required interval. The current Surveillance Requirement can increase the potential for a plant transient.

B. Control Rod Block Instrumentation Surveillance Requirements

1. The Rod Block Monitor (RBM) channel functional tests are required to be performed quarterly in Operational Condition 1 with thermal power greater than or equal to 30% of rated thermal power. The tests are also currently required to be performed within 24 hours prior to exceeding 30% of rated thermal power, if not performed within the previous 7 days. Exceeding 30% of rated thermal power does not affect the ability of the RBM to perform its function. The quarterly test frequency provides sufficient assurance the RBM is functioning properly. Therefore, the requirement to perform the functional tests prior to exceeding 30% of rated thermal power causes the diversion of plant personnel and resources for unnecessary testing without a commensurate increase in plant safety.

2. In addition to the periodic surveillance test frequencies specified in TS Table 4.3.6-1, the channel functional tests for the APRM, SRM, IRM and Reactor Coolant System Recirculation Flow control rod block instrumentation trip functions are currently required to be performed within 24 hours prior to startup, if not performed within the previous 7 days. Plant startup does not affect the ability of the control rod block instrumentation trip functions to perform as required. The periodic test frequencies provide sufficient assurance the control rod block instrumentation is functioning properly. Therefore, the requirement to perform the functional tests prior to startup causes the diversion of plant personnel and resources for unnecessary testing without a commensurate increase in plant safety.

3. The Reactor Mode Switch Shutdown Position control rod withdrawal block is required to be operable in Operational Conditions 3 and 4. Testing of this interlock with the reactor mode switch in any other position than Shutdown cannot be performed without using jumpers or lifted leads. Since SR 4.0.4 prohibits the planned entry into an Operational Condition or other specified applicable condition unless the Surveillance Requirement(s)

associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified, a planned entry into Operational Condition 3 from Operational Condition 1 or 2 would be prohibited by TS if the functional test had not been performed within the required interval.

C. Source Range Monitor Surveillance Requirements

SR 4.3.7.6.b requires the SRM channel functional test to be performed at least once per 31 days in Operational Condition 2 with IRMs on range 2 or below and in Operational Conditions 3 and 4. The channel functional test is also required to be performed within 24 hours prior to moving the reactor mode switch from the "Shutdown" position, if not performed within the previous 7 days. SR 4.9.2.b requires the SRM channel functional test to be performed at least once per 7 days in Operational Condition 5 and within 24 hours prior to the start of core alterations.

Moving the reactor mode switch from the "Shutdown" position or starting core alterations does not affect the ability of the SRMs to perform their required function. The requirements to perform the functional tests prior to moving the reactor mode switch or starting core alterations can cause unnecessary testing to be performed.

D. Power Distribution Limits Surveillance Requirements

Power distribution limits are required to be verified at least once per 24 hours in Operational Condition 1, when thermal power is greater than or equal to 25% of rated thermal power. Power distribution limits are also currently required to be verified within 12 hours after completion of a thermal power increase of at least 15% of rated thermal power. The additional requirement can create confusion as to how often the Surveillance is required (i.e., after every 15% power increase, or at the end of any single power increase greater than 15%).

4. TECHNICAL ANALYSIS

A. Reactor Protection System Instrumentation Surveillance Requirements

1. TS Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," would be revised to delete the requirements to perform IRM and APRM channel checks and channel functional tests prior to startup.

   Performing a reactor startup does not affect the ability of the APRMs and IRMs to perform their function. The normal surveillance test frequencies continue to provide sufficient assurance the APRMs and IRMs are functioning properly. SR 4.0.4 continues to require the Surveillance Requirement(s) associated with each Limiting Condition for Operation to be performed within

the applicable surveillance interval before entry into an Operational Condition or other specified applicable condition.

The proposed change will reduce the burden on plant personnel preparing for plant startup by minimizing unnecessary testing while continuing to ensure the APRMs and IRMs are capable of performing their specified functions.

The proposed changes are consistent with changes previously approved by the NRC for the Browns Ferry Nuclear Plant, Units 1, 2 and 3 (Reference 2) and for the Limerick Generating Station, Units 1 and 2 (Reference 3). The proposed changes are also consistent with NUREG-1433.

2. TS Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," would be revised to allow performance of the APRM Neutron Flux - Upscale, Setdown channel functional test to be deferred for up to 12 hours after entry into Operational Condition 2 from Operational Condition 1.

   The proposed change would eliminate the requirement to use jumpers or lifted leads to perform the functional test prior to a planned entry into Operational Condition 2 in the case where the surveillance had not been performed within the applicable surveillance interval. The 12 hour allowance provides a reasonable time in which to complete the SR under conditions in which the potential for a plant transient is reduced.

   The proposed changes are consistent with changes previously approved by the NRC for the Browns Ferry Nuclear Plant, Units 1, 2 and 3 (Reference 2) and the Limerick Generating Station, Units 1 (Reference 4) and 2 (Reference 5). The proposed changes are also consistent with NUREG-1433.

B. <u>Control Rod Block Instrumentation Surveillance Requirements</u>

   1. TS Table 4.3.6-1, "Control Rod Block Instrumentation Surveillance Requirements," would be revised to delete the requirement to perform the RBM channel functional test during startup, prior to exceeding 30% of RTP.

      Exceeding 30% of rated thermal power does not affect the ability of the RBM to perform its function. The normal quarterly surveillance test frequency continues to provide sufficient assurance the RBM is functioning properly. SR 4.0.4 continues to require the Surveillance Requirement(s) associated with each Limiting Condition for Operation to be performed within the applicable surveillance

interval before entry into an Operational Condition or other specified applicable condition.

The proposed change will reduce the burden on plant personnel performing plant startup by minimizing unnecessary testing while continuing to ensure the RBM is capable of performing its specified function.

The proposed change is consistent with changes previously approved by the NRC for the Limerick Generating Station, Units 1 and 2 (Reference 6). The proposed change is also consistent with NUREG-1433.

2. TS Table 4.3.6-1, "Control Rod Block Instrumentation Surveillance Requirements," would be revised to delete the requirements to perform channel functional tests prior to startup for the APRM, SRM, IRM and Reactor Coolant System Recirculation Flow control rod block instrumentation trip functions.

   Plant startup does not affect the ability of the control rod block instrumentation trip functions to perform as required. The periodic test frequencies provide sufficient assurance the control rod block instrumentation is functioning properly. SR 4.0.4 continues to require the Surveillance Requirement(s) associated with each Limiting Condition for Operation to be performed within the applicable surveillance interval before entry into an Operational Condition or other specified applicable condition.

   The proposed changes will reduce the burden on plant personnel preparing for plant startup by minimizing unnecessary testing while continuing to ensure the control rod block instrumentation is capable of performing its specified function.

   The proposed changes are consistent with changes previously approved by the NRC for the Limerick Generating Station, Units 1 and 2 (Reference 6).

3. TS Table 4.3.6-1, "Control Rod Block Instrumentation Surveillance Requirements," would be revised to allow performance of the Reactor Mode Switch Shutdown Position channel functional test to be deferred until 1 hour after the reactor mode switch is in the shutdown position.

   The proposed change would eliminate the requirement to use jumpers or lifted leads to perform the functional test prior to a

planned shutdown in the case where the surveillance had not been performed within the applicable surveillance interval.

The proposed change is consistent with changes previously approved by the NRC for the Limerick Generating Station, Units 1 and 2 (Reference 6). The proposed change is also consistent with NUREG-1433.

C. Source Range Monitor Surveillance Requirements

SR 4.3.7.6.b would be revised to delete the requirement to perform the source range monitor channel functional test prior to moving the reactor mode switch from the Shutdown position. SR 4.9.2.b would be revised to delete the requirement to perform the source range monitor channel functional test prior to the start of core alterations.

Moving the reactor mode switch from the "Shutdown" position or starting core alterations does not affect the ability of the SRMs to perform their required function. The periodic test frequencies provide sufficient assurance the SRMs are functioning properly. SR 4.0.4 continues to require the Surveillance Requirement(s) associated with each Limiting Condition for Operation to be performed within the applicable surveillance interval before entry into an Operational Condition or other specified applicable condition.

The proposed changes will reduce the operational burden on plant personnel by minimizing unnecessary testing while continuing to ensure the SRMs are capable of performing their specified function.

The proposed changes are consistent with changes previously approved by the NRC for the Limerick Generating Station, Units 1 and 2 (Reference 6). The proposed changes are also consistent with NUREG-1433.

D. Power Distribution Limits Surveillance Requirements

SRs 4.2.1, 4.2.3 and 4.2.4 would be revised to require verification of power distribution limits within 12 hours after core thermal power is greater than or equal to 25% of rated thermal power (RTP) and at least once every 24 hours thereafter. The requirements to verify power distribution limits within 12 hours after completion of a thermal power increase of at least 15% of RTP would be deleted. The specific exemptions from the provisions of SR 4.0.4 would be deleted.

Verifying power distribution limits within 12 hours of reaching or exceeding 25% of RTP will typically require the surveillances to be performed sooner than under the current Surveillance Requirements. A single verification during plant startup is sufficient given the large margins to operating limits

at low power levels. After the initial verification, the surveillances would be performed every 24 hours to identify adverse trends. The 24 hour interval is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation.

The exceptions to SR 4.0.4 are no longer required since the new SRs provide an allowance to complete the power distribution limits verifications after core thermal power is greater than or equal to 25%.

The proposed changes are consistent with NUREG-1433.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

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

   Response: No.

   The proposed amendment would revise the Technical Specification (TS) Surveillance Requirements (SRs) for certain Reactor Protection System and Control Rod Block Instrumentation, the source range monitors and power distribution limits, consistent with NUREG-1433, "Standard Technical Specifications (STS) General Electric Plants, BWR/4," Revision 2. No changes are being made to any instrumentation setpoints or plant components. The revised SRs continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met.

   Since the proposed changes do not affect any accident initiator and since the associated equipment will remain capable of performing its design function, the proposed change does not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not change the design function or operation of any plant equipment. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

No changes are being made to any plant instrumentation setpoints or to the required level of redundancy. No changes are being made to any power distribution limits.

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

Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50 Appendix A, General Design Criterion (GDC) 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

10 CFR 50 Appendix A, GDC 13 requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

10 CFR 50 Appendix A, GDC 20 requires that protection systems shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

10 CFR 50 Appendix A, GDC 21 requires that protection systems shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

The changes to Surveillance Requirements proposed in this license amendment request will minimize unnecessary testing while continuing to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

PSEG has determined the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or a surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7. REFERENCES

1. NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4," Revision 2.

2. Issuance of Technical Specification Amendments for the Browns Ferry Nuclear Plants Units 1, 2, and 3 (TAC Nos. M91270, M91271 and M91272, November 2, 1995

3. Limerick Generating Station, Units 1 and 2 (TAC Nos. M93216 and M93217), February 14, 1996

4. Limerick Generating Station, Unit 1 - Issuance of Amendment Re: Power Range Neutron Monitoring (TAC No. MA6965), April 12, 2000

5. Limerick Generating Station, Unit 2 - Issuance of Amendment Re: Power Range Neutron Monitoring (TAC No. MA6966), January 16, 2001

6. License Amendment on Optional Scram Insertion Time Testing and Control Rod Block and Source Range Monitoring Instrumentation, Limerick Generating Station, Units 1 and 2 (TAC Nos. M90377, M90378, M90508 and M90509), July 18, 1995

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

Technical Specification	Page
3/4.2.1	3/4 2-1
3/4.2.3	3/4 2-3
3/4.2.4	3/4 2-5
Table 4.3.1.1-1	3/4 3-7 3/4 3-8
Table 4.3.6-1	3/4 3-60 3/4 3-61
3/4.3.7	3/4 3-88
3/4.9.2	3/4 9-4
Bases 3/4.3.1	B 3/4 3-1
Bases 3/4.3.6	B 3/4 3-4

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

~~a. At least once per 24 hours,~~

~~b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and~~

b. ~~c.~~ Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

~~d. The provisions of Specification 4.0.4 are not applicable.~~

a. Once within 12 hours after THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER and at least once per 24 hours thereafter.

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within 1 hour, MCPR is determined to be greater than or equal to the EOC-RPT inoperable limit specified in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT:

~~a. At least once per 24 hours,~~

~~b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and~~

b. ~~c.~~ Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

~~d. The provisions of Specification 4.0.4 are not applicable.~~

a. Once within 12 hours after THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER and at least once per 24 hours thereafter.

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit specified in the CORE OPERATING LIMITS REPORT:

~~a. At least once per 24 hours,~~

~~b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and~~

b. ~~c.~~ Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

~~d. The provisions of Specification 4.0.4 are not applicable.~~

a. Once within 12 hours after THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER and at least once per 24 hours thereafter.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	~~S/U[b]~~, S S	~~S/U[c]~~, W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	~~S/U[b]~~, S S	~~S/U[c]~~, W W	SA SA	2 3, 4, 5
b. Flow Biased Simulated Thermal Power - Upscale	S, D[g]	~~S/U[c]~~, Q	W[d][e], SA, R[h]	1
c. Fixed Neutron Flux - Upscale	S	~~S/U[c]~~, Q	W[d], SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q[k]	R	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	Q[k]	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. This item intentionally blank				
7. Drywell Pressure - High	S	Q[k]	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5(j)
b. Level Transmitter/Trip Unit	S	Q(k)	R	1, 2, 5(j)
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the ~~previous 7 days.~~

(c) ~~Within 24 hours prior to startup, if not performed within the previous 7 days.~~ DELETED

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).

(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.

(i) This item intentionally blank

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(k) Verify the tripset point of the trip unit at least once per 92 days.

(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	~~S/U(c)(d),~~ Q(c)	SA	1*
b. Inoperative	NA	~~S/U(c)(d),~~ Q(c)	NA	1*
c. Downscale	NA	~~S/U(c)(d),~~ Q(c)	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	NA	~~S/U(b)~~ Q	SA	[illegible]
b. Inoperative	NA	~~S/U(b)~~ Q	NA	1, 2, 5
c. Downscale	NA	~~S/U(b)~~ Q	SA	1
d. Neutron Flux - Upscale, Startup	NA	~~S/U(b)~~ Q	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	~~S/U(b)~~ W	NA	2, 5
b. Upscale	NA	~~S/U(b)~~ W	R	2, 5
c. Inoperative	NA	~~S/U(b)~~ W	NA	2, 5
d. Downscale	NA	~~S/U(b)~~ W	R	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	~~S/U(b)~~ W	NA	2, 5
b. Upscale	NA	~~S/U(b)~~ W	R	2, 5
c. Inoperative	NA	~~S/U(b)~~ W	NA	2, 5
d. Downscale	NA	~~S/U(b)~~ W	R	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High (Float Switch)	NA	Q	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	~~S/U(b)~~ Q	SA	1
b. Inoperative	NA	~~S/U(b)~~ Q	NA	1
c. Comparator	NA	~~S/U(b)~~ Q	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R (e)	NA	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. ~~Within 24 hours prior to startup, if not performed within the previous 7 days.~~ DELETED

c. Includes reactor manual control multiplexing system input.

d. ~~Within 24 hours prior to exceeding 30% of RATED THERMAL POWER, if not performed within the previous 7 days.~~ DELETED

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

e. Not required to be performed until 1 hour after reactor mode switch is in the shutdown position.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK at least once per:

   a) 12 hours in CONDITION 2*, and

   b) 24 hours in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. ~~Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and~~

2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3 cps with the detector fully inserted.

d. The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2* or 3 from OPERATIONAL CONDITION 1.

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

b. Performance of a CHANNEL FUNCTIONAL TEST:

~~1. Within 24 hours prior to the start of CORE ALTERATIONS, and~~

~~2. At least once~~ at per 7 days.

c. Verifying ~~that the~~ channel count rate is at least 3 cps.

1. Prior to control ~~rod~~ withdrawal,

2. Prior to and at least once per 12 hours during CORE ALTERATIONS***, and

3. At least once per 24 hours***.

d. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, verifying that the RPS circuitry "shorting links" have been removed, within 8 hours prior to and at least once per 12 hours during the time any control rod is withdrawn.**

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.
***Except as noted in Specifications 3.9.2.d and 3.9.2.e.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter to T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times. Selected sensor response time testing requirements were eliminated based upon NEDO-32291, "System Analyses for Elimination of Selected Response Time Testing Requirements," as approved by the NRC and documented in the SER (letter to R.A. Pinelli from Bruce A. Boger, dated December 28, 1994). The Reactor Protection System Response Times are located in UFSAR Table 7.2-3.

INSERT A

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2 and GENE-770-06-2-A. "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications." The safety evaluation reports documenting NRC approval of NEDC-30936P-A and GENE-770-06-2-A are contained in letters to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1), D. N. Grace to C. E. Rossi dated December 9, 1988 (Part 2), and G. J. Beck from C. E. Rossi dated September 13, 1991.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. INSERT B

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63 and 64.

3/4.3.7.2 DELETED

3/4.3.7.3 DELETED

Insert A

As noted, the SR for the APRM Neutron Flux - Upscale, Setdown channel functional test is not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1, since testing of the OPERATIONAL CONDITION 2 required APRM Function cannot be performed in OPERATIONAL CONDITION 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into OPERATIONAL CONDITION 2 if the 7 day frequency is not met per SR 4.0.2. In this event, the SR must be performed within 12 hours after entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

Insert B

As noted, the SR for the Reactor Mode Switch Shutdown Position functional test is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into OPERATIONAL CONDITIONS 3 and 4 if the 18 month frequency is not met per SR 4.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.