Public Service
Electric and Gas
Company

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Vice President - Nuclear Operations

OCT 1 8 1993

NLR-N93015
LCR 93-03

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

LICENSE AMENDMENT APPLICATION
STI/AOT EXTENSIONS FOR ISOLATION ACTUATION INSTRUMENTATION
FACILITY OPERATING LICENSE NPF-57
HOPE CREEK GENERATING STATION
DOCKET NO. 50-354

This letter submits an application for amendment to Appendix A of Facility Operating License NPF-57 for the Hope Creek Generating Station and is being filed in accordance with 10CFR50.90. The changes that are proposed in this submittal would extend the surveillance test intervals (STIs) and allowed out-of-service times (AOTs) for the isolation actuation instrumentation. Attachment 1 contains a detailed description of and justification for the proposed changes. Based upon the justification provided, PSE&G believes that the proposed changes do not involve a significant hazards consideration pursuant to 10CFR50.92. The technical information contained in Attachment 1 is based upon General Electric Company (GE) Licensing Topical Report (LTR) NEDC-30851P-A, Supplement 2 (March 1989) and GE LTR NEDC-31677P-A (July 1990). As these changes reflect NRC approved, generic changes contained in the LTRs, PSE&G believes that a detailed NRC branch review or specialist review should not be required.

Attachment 2 contains marked up Technical Specification pages which reflect the proposed changes.

Upon NRC approval, please issue a license amendment which will be effective upon issuance and shall be implemented within 60 days of issuance. This latitude permits appropriate procedural modifications necessary to implement the proposed changes.

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Should you have any questions or comments on this submittal, please do not hesitate to contact us.

Sincerely,

Affidavit
Attachments (2)

C Mr. T. T. Martin, Administrator - Region I
U. S. Nuclear Regulatory Commission
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King of Prussia, PA 19406

Mr. S. Dembek, Licensing Project Manager
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Mr. C. S. Marschall (S09)
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REF: NLR-N93015

LCR 93-03

STATE OF NEW JERSEY)

) SS.

COUNTY OF SALEM)

J. J. Hagan, being duly sworn according to law deposes and says:

I am Vice President - Nuclear Operations of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning the Hope Creek Generating Station, are true to the best of my knowledge, information and belief.

JJHagan

Subscribed and Sworn to before me
this 18th day of October, 1993

Kimberly Jo Brown

Notary Public of New Jersey

My Commission expires on KIMBERLY JO BROWN NOTARY PUBLIC OF NEW JERSEY My Commission Expires April 21, 1998

ATTACHMENT 1
PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

LICENSE AMENDMENT APPLICATION
STI/AOT EXTENSIONS FOR ISOLATION ACTUATION INSTRUMENTATION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57 NLR-N93015
DOCKET NO. 50-354 LCR 93-03

I. DESCRIPTION OF THE PROPOSED CHANGES

This license amendment application proposes to change Technical Specification (TS) 3/4.3.2, "ISOLATION ACTUATION INSTRUMENTATION" and its associated Bases, such that:

A. The allowed out-of-service time (AOT) for surveillance testing specified in Note (a) to Table 3.3.2-1 is extended from 2 to 6 hours.

B. The AOTs for maintenance are extended to 12 hours for isolation instrumentation common to reactor protection system (RPS) instrumentation and to 24 hours for isolation instrumentation not common to RPS instrumentation. The AOT remains one hour for situations in which there is a loss of function and there are no operable channels for the given trip function. A footnote is added to Table 3.3.2-1 to identify the trip functions with instrumentation common to RPS instrumentation.

C. The test frequency in Footnotes (a) and (b) of Table 4.3.3.1-1 are changed from once per 31 days to once per 92 days.

D. The channel functional test requirements specified in Table 4.3.3.1-1 are extended from monthly to quarterly for the following trip functions:

1. Primary Containment Isolation:

   a. Reactor Vessel Water Level -
      1) Low Low, Level 2
      2) Low Low Low, Level 1
   b. Drywell Pressure - High
   c. Reactor Building Exhaust Radiation - High
   d. Manual Initiation

2. Secondary Containment Isolation:

   a. Reactor Vessel Water Level - Low Low, Level 2
   b. Drywell Pressure - High
   c. Refuel Floor Building Exhaust Radiation - High
   d. Reactor Building Exhaust Radiation - High
   e. Manual Initiation

3. Main Steam Line Isolation:

   a. Reactor Vessel Water Level - Low Low Low, Level 1
   b. Main Steam Line Radiation - High, High
   c. Main Steam Line Pressure - Low
   d. Main Steam Line Flow - High
   e. Condenser Vacuum - Low
   f. Main Steam Line Tunnel Temperature - High
   g. Manual Initiation

4. Reactor Water Cleanup System Isolation:

   a. RWCU Delta Flow - High
   b. RWCU Delta Flow - High, Timer
   c. RWCU Area Temperature - High
   d. RWCU Area Ventilation Delta Temperature - High
   e. SLCS Initiation
   f. Reactor Vessel Water Level - Low Low. Level 2
   g. Manual Initiation

5. Reactor Core Isolation Cooling System Isolation:

   a. RCIC Steam Line Delta Pressure (Flow) - High
   b. RCIC Steam Line Delta Pressure (Flow) - High, Timer
   c. RCIC Steam Supply Pressure - Low
   d. RCIC Turbine Exhaust Diaphragm Pressure - High
   e. RCIC Pump Room Temperature - High
   f. RCIC Pump Room Ventilation Ducts Delta Temperature - High
   g. RCIC Pipe Routing Area Temperature - High
   h. RCIC Torus Compartment Temperature - High
   i. Drywell Pressure - High

6. High Pressure Coolant Injection System Isolation:

   a. HPCI Steam Line Delta Pressure (Flow) - High
   b. HPCI Steam Line Delta Pressure (Flow) - High, Timer
   c. RCIC Steam Supply Pressure - Low
   d. HPCI Turbine Exhaust Diaphragm Pressure - High
   e. HPCI Pump Room Temperature - High
   f. HPCI Pump Room Ventilation Ducts Delta Temperature - High
   g. HPCI Pipe Routing Area Temperature - High
   h. HPCI Torus Compartment Temperature - High
   i. Drywell Pressure - High

7. RHR System Shutdown Cooling Mode Isolation:

   a. Reactor Vessel Water Level - Low, Level 3
   b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High
   c. Manual Initiation

E. Bases Section 3/4.3.2 is revised to reference the General Electric (GE) Licensing Topical Reports (LTRs) which justify the above proposed changes to the isolation actuation instrumentation.

The proposed TS changes described above are consistent with the changes proposed and approved in the referenced GE documents and associated NRC safety evaluation reports (References 1 - 4) with the exception that the subsequent enhancements included in Reference 5 have been adopted, the loss of function issue has been addressed, and a footnote has been added to identify which trip functions have instrumentation common to RPS.

II. REASON FOR THE PROPOSED CHANGES

The technical assessment of the proposed changes contained in the GE LTRs indicates a positive benefit and net improvement in overall plant safety and operation. This conclusion was based upon consideration of the impact of the changes on the isolation failure frequency as well as impact of the following factors:

A. Potential for inadvertent scrams

B. Excessive test actuation cycles (equipment wearout)

C. Diversion of plant personnel

D. Potential for test-caused failures

E. Potential increased risk from shutdown due to limiting conditions of operation

Contributing factors A through E are referenced in Section 5.6 of Reference 2 and their impact on plant safety is discussed in detail in Section 4.2 of Reference 6.

III. JUSTIFICATION FOR THE PROPOSED CHANGES

The generic GE analyses contained in References 1 and 2 evaluated the effect of the proposed changes to the STIs and AOTs for the isolation actuation instrumentation and demonstrated that the isolation failure frequency (IFF) is not significantly affected by the proposed changes. The calculated change in IFF for the proposed changes meets the established acceptance criterion for ensuring negligible change to the IFF. Furthermore, when the factors described in Section II of this submittal are considered, in addition to the IFF, the overall effect on plant safety is judged to be an improvement.

References 3 and 4 concluded that the associated GE reports provide an acceptable basis for extending STIs and AOTs for isolation actuation instrumentation; however, these NRC safety evaluation reports (SERs) also required that two issues be addressed to justify the applicability of the generic analysis to individual plants when specific facility Technical Specifications are considered for revision. These issues were: 1) confirmation of the applicability of the generic analyses to the specific plant and 2) confirmation that any increase in instrument drift due to the extended STIs is properly accounted for in the setpoint calculation methodology. The following discussion provides information to address these issues.

A. Confirmation of the Applicability of the Generic Analyses

Our confirmation of the applicability of the generic analyses to Hope Creek is based upon the following:

1. Appendix A of Reference 1 and Appendix E of Reference 2 identify Public Service Electric and Gas Company/Hope Creek as a participating utility/plant in the Isolation Actuation Technical Specification Improvement Analysis. PSE&G has maintained its participation and involvement on the BWR Owners Group Technical Specification Improvement Committees thereby assuring that the development of these generic reports encompass the Hope Creek Generating Station.

2. PSE&G has reviewed the applicable GE LTRs and has verified their applicability to the Hope Creek Generating Station.

B. Confirmation that Instrument Drift is Properly Considered

The NRC staff has provided guidance on addressing the issue of instrument drift in Reference 7. This guidance indicated that:

> "...licensees need only confirm that the setpoint drift which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS and ESFAS instrument setpoint calculation or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift."

In order to satisfy this requirement, PSE&G applied a two fold approach to the issue of instrument drift. This two fold approach involved the following:

1. The setpoint calculations for all instrumentation affected by the changes proposed in this amendment application were reviewed. Results of this review indicate that, in all cases, the loop (setpoint) drift calculation was based on an eighteen month interval; therefore, the proposed STI extensions from monthly to quarterly are well bounded by the existing setpoint calculations.

2. The surveillance tests for the NUMAC instrumentation (used for Trip Functions 3.f, 4.a - d, 5.e - h, and 6.e - h) and the radiation monitoring system examine digital components in the instrument loop. Since these components are digital, there is no associated inherent drift. The drift for the component will therefore always be zero, independent of the interval at which it is tested.

   For the remaining analog instrumentation, data for each trip unit consisted of the "as found" and "as left" trip setpoint settings over a twelve month period. The actual observed drift over the twelve month period, in all cases, was found to be conservatively bounded by the total loop allowance for a six month period. The results of this evaluation are documented in References 8 and 9 and are available for NRC staff review.

IV. SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

PSE&G has, pursuant to 10 CFR 50.92, reviewed the proposed amendment to determine whether the request involves a significant hazards consideration. We have determined that operation of the Hope Creek Generating Station in accordance with the proposed changes:

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

   The proposed changes to the isolation actuation instrumentation were judged to potentially affect plant safety through their impact on the isolation failure frequency (IFF). The generic analyses contained in Licensing Topical Report (LTR) NEDC-30851P-A, Supplement 2 and LTR NEDC-31677P-A assessed the impact of changing the isolation actuation instrumentation surveillance test intervals (STIs) and allowed out-of-service times (AOTs) on the IFF. The analyses contained in these LTRs demonstrate that the proposed changes have a negligible effect on the IFF, and when all contributing factors are considered, the net impact of the proposed changes is to improve plant safety. These generic analyses have been verified to be applicable to the HCGS as indicated in Section III above. Since the proposed changes do not significantly affect the IFF and have a beneficial impact on plant safety when all factors are considered, the proposed changes will not significantly increase the probability or consequences of a previously analyzed accident.

2. **Will not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   Increasing the AOTs and STIs for the isolation actuation instrumentation does not alter the function of the equipment performing the isolation functions nor involve any type of plant modification. Additionally, no new modes of plant operation are involved with these changes. The proposed changes therefore will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Will not involve a significant reduction in a margin of safety.**

   The proposed changes to the isolation actuation instrumentation were judged to potentially affect plant safety through their impact on the IFF. As requested by the BWR Owners' Group, GE performed analyses to evaluate the effect of the proposed changes on the IFF. The NRC staff has reviewed and approved the generic study contained in LTRs NEDC-30851P-A, Supplement 2 and NEDC-31677P-A and has concurred with the BWR Owners Group that the proposed changes do not significantly affect the IFF. Furthermore, the overall level of plant safety will be improved by the proposed changes. It can therefore be concluded that the proposed changes will not significantly reduce a margin of safety.

V. CONCLUSION

As discussed above, PSE&G has concluded that the proposed changes to the Technical Specifications do not involve a significant hazards consideration since the changes: (i) do not involve a significant increase in the probability or consequences of an accident previously evaluated, (ii) do not create the possibility of a new or different kind of accident from any accident previously evaluated, and (iii) do not involve a significant reduction in a margin of safety.

VI. REFERENCES

1. NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation Common to RPS and ECCS Instrumentation", dated March 1989

2. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", dated July 1990

3. Letter to D. N. Grace from C. E. Rossi dated January 6, 1989 (transmits NRC safety evaluation report for NEDC-30851P-A, Supplement 2)

4. Letter to S. D. Floyd from C. E. Rossi dated June 18, 1990 (transmits NRC safety evaluation report for NEDC-31677P-A)

5. GE Document No. OG90-579-32A, letter from W. P. Sullivan (GE) to USNRC, "Implementation Enhancements to Technical Specification Changes Given in Isolation Actuation Instrumentation Analysis", dated June 25, 1990

6. NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation) Part 2", dated December 1988

7. Letter from C. E. Rossi (NRR) to R. F. Janecek (BWROG), "Staff Guidance for Licensee Determination that the Drift Characteristics for Instrumentation Used in RPS Channels are Bounded by NEDC-30851P Assumptions When the Functional Test Interval is Extended from Monthly to Quarterly", dated April 27, 1988

8. PSE&G Internal Memorandum ELE-92-0667 from Robert Sandy, Hope Creek I&C, to C. Manges, Nuclear Licensing, dated December 9, 1992

9. PSE&G Internal Memorandum ELE-93-0489 from Robert Sandy, Hope Creek I&C, to L. Castagna, Nuclear Licensing, dated October 5, 1993

INSERT A

b. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for one trip system, either

1) place the inoperable channel(s) in the tripped condition within

a) 1 hour* for trip functions without an OPERABLE channel,

b) 12 hours for trip functions common to RPS instrumentation, and

c) 24 hours for trip functions not common to RPS instrumentation,

or

2) take the ACTION required by Table 3.3.2-1.

The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for both trip systems,

1) place the inoperable channel(s) in one trip system in the tripped condition within one hour

and

2) a) place the inoperable channel(s) in the remaining trip system in the tripped condition within

1) 1 hour for trip functions without an OPERABLE channel,

2) 12 hours for trip functions common tc RPS instrumentation, and

3) 24 hours for trip functions not common to RPS instrumentation,

or

b) take the ACTION required by Table 3.3.2-1.

The provisions of Specification 3.0.4 are not applicable.

INSERT B

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation Common to RPS and ECCS Instrumentation", and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation". The safety evaluation reports documenting NRC approval of NEDC-30851P-A, Supplement 2 and NEDC-31677P-A are contained in letters to D. N. Grace from C. E. Rossi dated January 6, 1989 and to S. D. Floyd from C. E. Rossi dated June 18, 1990.