UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated October 18, 1993, as supplemented by letter dated March 7, 1994, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Hope Creek Generating Station, Technical Specification (TS). The requested changes would revise TS 3/4.3.2 and its associated Bases to increase surveillance test intervals and add allowable out-of-service times for isolation actuation instrumentation. The licensee's amendment application stated the requested changes are consistent with the NRC staff's previous approvals of General Electric Company's (GE) Licensing Topical Reports (LTRs). The licensee's proposed changes to both surveillance test intervals and out-of-service times are also consistent with the guidance provided in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The March 7, 1994, letter provided clarifying information that did not change the initial no significant hazards consideration determination.

Specifically, the proposed changes are as follows:

A. The allowed out-of-service time (AOTs) for surveillance testing of Isolation Actuation Instrumentation would be 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. 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 Delay 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 GE LTRs which justify the above proposed changes to the isolation actuation instrumentation.

2.0 EVALUATION

The licensee has proposed changes to TS 3/4.3.2 "ISOLATION ACTUATION INSTRUMENTATION" and its associated Bases.

The proposed changes are based on the NRC staff's previous approvals of the following GE LTR'S:

a. NEDC-30851P-A, Supplement 2 "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," dated March 1989. This LTR was approved by letter and enclosed safety evaluation dated January 6, 1989, from C. E. Rossi (NRC) to D. N. Grace (BWR Owners Group).

b. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," dated July 1990. This LTR was approved by letter and enclosed safety evaluation dated June 18, 1990, from C. E. Rossi (NRC) to S. D. Floyd (BWR Owners Group).

c. NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation) Part 2," dated December 1988. This LTR was approved by letter and enclosed safety evaluation dated December 9, 1988, from C. E. Rossi (NRC) to D. N. Grace (BWR Owners Group).

d. 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 Function Test Interval is extended from Monthly to Quarterly", dated April 27, 1988.

The licensee has stated the reason for the proposed TS changes are as follows:

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 NEDC-31677P-A, Reference 2.b above, and their impact on plant safety is discussed in detail in Section 4.2 of NEDC-30936P-A (Reference 2.C above).

The licensee has justified the proposed changes as stated below. The generic GE analyses contained in References 2.0a and 2.0b evaluated the effect of the proposed changes to the surveillance test intervals (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 factors A thru E described above are considered, in addition to the IFF, the overall effect on plant safety is judged to be an improvement.

The NRC Safety Evaluations for NEDC-30851P-A and NEDC-31677P-A 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 are: 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 licensees confirmation of the applicability of the generic analyses to Hope Creek is based upon the following paraphrased from PSE&G's October 18, 1993 license amendment application:

1. Appendix A of Reference 2.0a and Appendix E of Reference 2.0b, and using above References, identify Public Service Electric and Gas Company (PSE&G)/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.

The licensees confirmation that Instrument Drift is properly considered is supported below.

The NRC staff has provided guidance on addressing the issue of instrument drift in the guidance provided in C. E. Rossi's April 27, 1988 letter to R. F. Janecek referenced above. 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 [reactor protection system] RPS and [engineered safety features actuation system] 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, the licensee stated they 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 18-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 [Nuclear Measurement Analysis and Control] NUMAC instrumentation (used for Trip Functions 3.f, 4.a-d, 5.e-h, and 6.3-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 12-month period. The actual observed drift over the 12-month period, in all cases, was found to be conservatively bounded by the total loop allowance for a 6-month period. The results of this evaluation are documented in References 8 and 9 of the licensee's October 18, 1993 application for license amendment.

The staff finds the licensees justification demonstrates that the proposed TS changes are in conformance with the standard technical specifications (STS) and related NRC guidance and are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements and a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 64615). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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.

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