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January 25, 1991

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Quad Cities Nuclear Power Station Units 1 and 2
Request for Exemption of the Feedwater Check
Valves from 10 CFR 50 Appendix J Test Program
NRC Docket Nos. 50-254 and 50-265

Dr. Murley:

On September 25, 1990 members of the Commonwealth Edison's and Nuclear Regulatory Commission's (including Region III and NRR representatives) staffs conducted a meeting to discuss Commonwealth Edison's actions to improve the performance of containment valves during 10 CFR 50 Appendix J testing at Quad Cities Station. During that meeting, Commonwealth Edison proposed to submit a request to exempt the feedwater check valves from the Appendix J Test Program. The basis for this request is a calculation, which demonstrates that water remains in the feedwater line following a loss-of-coolant accident. During the September 25, 1990 meeting, Commonwealth Edison provided an overview of Quad Cities Station's request. The NRR representative indicated that the request was reasonable and should be formally submitted.

In lieu of performing an air test of the feedwater check valves, Commonwealth Edison proposes to perform a water leakage test of the valves to ensure water remains in the piping between the reactor vessel and feedwater valves. This request does not involve a change to the Technical Specifications or Final Safety Analysis Report.

If there are any questions or comments regarding this submittal, please contact me at 708/515-7283.

Very truly yours,

Rita Stols

Rita Stols
Nuclear Licensing Administrator

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Attachments

A: Request for Exemption of the Feedwater Check Valves from the requirements of 10CFR50 Appendix J

B: Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments 164 and 101 to Facility Operating Licenses DPR-57 and NPF-5.

C: Feedwater Line Thermal Hydraulic Behavior during LOCA conditions for Quad Cities Nuclear Power Station, RSA-Q-90-03 dated November 21, 1990.

Figures

1: Feedwater Piping Inside the Drywell

2: Feedwater Piping Outside the Drywell

3: Condensate and Feedwater System

4: FSAR Figure 5.2.17 "Containment Pressure Response Following Design Basis Loss-of-Coolant Accident"

cc: A.B. Davis, Regional Administrator, RIII
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ATTACHMENT A

QUAD CITIES NUCLEAR POWER STATION

REQUEST FOR EXEMPTION OF THE FEEDWATER CHECK VALVES

FROM 10 CFR 50 APPENDIX J TEST REQUIREMENTS

Background

On June 15, 1990, Region III issued the results of an inspection (50-254/89024; 50-265/89024) which was performed on the Quad Cities Station 10CFR50 Appendix J Test Program. As a result of the inspection, a Notice of Violation, which cited ineffective corrective actions to repetitive Appendix J test failures, was issued. Commonwealth Edison provided a response to the violation in a letter dated August 17, 1990. Subsequently, Region III requested that a meeting be conducted to discuss the response. On September 25, 1990, members of the Commonwealth Edison Company's and the Nuclear Regulatory Commission's (including representatives from both Region III and Nuclear Reactor Regulation) staffs conducted that meeting.

During the September 25, 1990 meeting, Commonwealth Edison provided an update of the investigations which are underway to resolve the leakage pathway concerns. As a result of these investigations, Commonwealth Edison identified that Georgia Power's Plant Hatch requested and received NRC's approval for eliminating the feedwater check valves from the Plant Hatch's 10 CFR 50 Appendix J test program. The basis for the NRC's approval to remove the valves from the Appendix J test program was that water was present in the feedwater lines during post accident conditions. At the time of the meeting, preliminary calculations confirmed that water would also be present in the feedwater lines following a loss-of-coolant accident (LOCA) at Quad Cities. Commonwealth Edison proposed to submit a similar request for Quad Cities Station and provided a brief overview of the technical justification for the request. The NRR representative indicated that the request was reasonable and should be formally submitted.

Plant Hatch Request

The basis for eliminating the feedwater check valves from the 10 CFR 50 Appendix J Test Program at Plant Hatch was an analysis that demonstrated that water would remain in the feedwater lines between the vessel and inboard valve following a loss-of-coolant accident (LOCA). In lieu of the Appendix J test, a stringent water leakage test is required to ensure that the lines are water filled following the LOCA. Since the lines are no longer potential air leakage pathways, the leakage is excluded from the 0.6 and 0.75 La air leakage totals. A copy of the NRC's Safety Evaluation Report is contained in Attachment B for the Staff's reference.

Quad Cities Station Request

10 CFR 50 Appendix J, Section II.H defines the valves which require Type C tests to include those that are in the feedwater piping. Commonwealth Edison requests the NRC's approval to discontinue Type C testing of the feedwater check valves on the basis that water remains in the feedwater line following the most limiting blowdown/fluid carryover event. Commonwealth Edison proposes to perform a water leakage test in lieu of the air test to assure water is maintained in the line.

Commonwealth Edison has performed an analysis for Quad Cities Station which examines the reactor vessel depressurization event and resultant feedwater line blowdown. The event was analyzed by the use of the computer code RELAP 5. A full six equation, two-phase calculation was performed to determine the amount of water which would remain in the feedwater line following reactor depressurization. Blowdown rates corresponding to a range of events, which included a small line break up to and including the design basis accident (DBA) of a recirculation system line break, were modeled. The analysis is contained in Attachment C.

Based on these calculations, the limiting event for feedwater system fluid carryover, which results in the least amount of water in the feedwater line, was the DBA recirculation line break. For this event, approximately 101 gallons of fluid remains in the feedwater line between the reactor vessel and the inboard feedwater check valve following reactor depressurization. With the presence of water in essentially a vertical run of the feedwater line between the vessel and inboard check valve, it is not possible for any containment atmosphere to escape. A schematic drawing of the feedwater line inside containment is provided as Figure 1.

In addition, a containment isolation function is also provided by a water seal which is created by a 15 foot elevation water head located between the heater outlet and the outboard feedwater check valve. (Reference Figure 2) The elevation head exerts approximately 6.2 psi pressure on the outboard feedwater check valve. The feedwater line, therefore, remains sealed provided that containment pressure does not exceed 6.2 psi.

As indicated in FSAR Figure 5.2.17 "Containment Pressure Response Following Design Basis Loss of Coolant Accident", containment pressure decays below 6.2 psi in approximately 55.6 hours following the initial reactor vessel blowdown. (Figure 5.2.17 is attached as Figure 4) Commonwealth Edison, therefore, proposes a minimum path water leakage rate test acceptance criteria for the inboard and outboard feedwater check valves of 1.82 gallons per hour to ensure that water remains in the feedwater line for 55.6 hours following blowdown. Following the 55.6 hours after reactor depressurization, the containment function will be performed by a water seal which is created by the 15 foot water head outside of containment.

Currently, the feedwater line which is located inside of containment is both seismic and safety-related piping. The feedwater piping, which is outside of containment, meets seismic and safety-related requirements up to the outboard feedwater check valve. The majority of the feedwater piping outside of containment, therefore, is non-seismic, non-safety related piping. If the NRC Staff approves this request to exempt the feedwater check valves from the Appendix J Test Program, Commonwealth Edison will upgrade this piping to meet seismic and safety-related piping requirements.

Commonwealth Edison has also considered the case of the feedwater line break as the initiator to a event. This event is classified as a medium break loss-of-coolant accident. Containment isolation provisions were not considered in this case due to the absence of resulting core damage.

In addition to the design features discussed above, other mitigating design features exist which support the removal of the feedwater check valves from the Appendix J Program:

1. A third check valve exists upstream of each outboard feedwater check valve. The valve (located on the "B" line) is water tested as part of the In-Service Test Program. No credit is taken for the third valve in the calculation to determine the amount of water which remains in the feedwater line following reactor vessel depressurization.

2. The High Pressure Coolant Injection (HPCI) System is injected into the vessel through one of the feedwater lines. HPCI takes its suction from the Contaminated Storage Tanks; therefore, the temperature of the coolant in the feedwater line would be reduced to less than 212 degrees Fahrenheit. The line, therefore, would remain filled with water.

3. The Reactor Core Injection Coolant (RCIC) System is injected into the vessel through the other feedwater line. The use of RCIC would assure that the line is water filled.

Conclusion

The Quad Cities Technical Specifications or the Final Safety Analysis Report does not contain a listing of valves which are required to be tested per 10 CFR 50 Appendix J. This request, therefore, does not require a change to either document. Commonwealth Edison requests the NRC's approval to exempt the feedwater check valves from the Appendix J Test Program.

If this request is approved:

1. Type C testing of the inboard and outboard feedwater check valves will continue until the outboard feedwater piping (up to the feedwater heaters) is upgraded to meet seismic and safety-related piping requirements. After the piping is upgraded, Type C tests would be discontinued on the valves.

2. Water leakage tests will be conducted (in lieu of the air test), at a pressure of Pa, on the inboard and outboard feedwater check valves. The acceptance criteria for this water leakage test will be established at 1.82 gallons per hour.

3. The requirement for inboard and outboard feedwater check valves water leakage test will be included in the In-Service Testing Program. The Program will be revised when the water leakage test commence on the feedwater check valves.

ATTACHMENT B

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 164 AND 101 TO

FACILITY OPERATING LICENSES DPR-57 AND NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated February 3, 1989, Georgia Power Company (the licensee) requested amendments to the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. Specifically, the proposed amendments would modify the TS for Units 1 and 2 to: (1) Change the maximum operating times for certain primary containment isolation valves (PCIVs) to account for a different method of measuring; (2) exclude several unit 1 containment penetrations and PCIVs from the local leak rate test (LLRT) program; (3) revise Unit 1 TS section 4.7.A.2 and Unit 2 TS section 4.6.1.3 to achieve similarity between the two documents, to comply with current 10 CFR 50 Appendix J testing requirements, and to specify an allowable leakage; (4) delete penetration 218A from Unit 1 TS Table 3.7-2; and (5) remove the isolation valves associated with the primary feedwater and the torus drainage and purification systems from Unit 2 TS section 3.6.1.2.

2.0 EVALUATION

2.1 Proposed Change 1 - Change the maximum operating times for certain primary containment isolation valves (PCIVs) to account for a different method of measuring.

The TS for Hatch Units 1 and 2 now contain table listings (Table 3.7-1 for Unit 1 and Table 3.6.3-1 for Unit 2) of power operated, automatically initiated PCIVs showing maximum operating times for isolating upon receipt of an appropriate signal. The operating times shown in the tables are based upon a "light-to-light" measuring method, which was used during the plant functional testing prior to reactor startup. However, the ASME Code, Section XI, requires that valve stroke times be measured from initiation of the actuating signal to the end of the actuating cycle, more commonly referred to as a "switch-to-light" measuring method. The changes to the maximum stroke times proposed by the licensee are merely to account for the change in stroke time measurement from the "light-to-light" method to the "swtich-to-light" method. Actual valve operation does not change and the new "switch-to-light" operating

times maintain the primary containment design bases as defined in the Unit 1 and Unit 2 Final Safety Analysis Reports. Since the actual valve operation remains the same, we find this proposed change to be acceptable.

2.2 Proposed Change 2 - Exclude Unit 1 valves 1E41-F021, F022, F040, and F049; valves 1E51-F001, F002, F028 and F040; and containment penetrations X-212, X-213, X-214 and X-215 from the Appendix J LLRT (Type C) program.

The named valves and the associated containment penetrations are those serving the reactor core isolation cooling (RCIC) turbine exhaust and turbine drain lines, and the high pressure coolant injection (HPCI) turbine exhaust and drain lines. These exhaust and drain lines terminate in the torus below the water level of the suppression pool. Since the torus water level is controlled within narrow limits at all times, continuous coverage of piping terminations is assured, thus providing a water seal between the atmospheres inside and outside the torus. Under this condition, Type C testing is not required by Appendix J. The licensee states, however, that leak rate testing of these valves will still be performed in accordance with Section XI of the ASME code as part of the inservice inspection (ISI) program.

The staff previously has approved the exemption of certain valves and associated penetrations from the Type C testing requirements based upon a similar argument that the piping involved terminated below the water level in the torus (Amendment #131, Hatch Unit 1, October 30, 1986; and Amendment #140, Hatch Unit 1, June 5, 1987).

Accordingly, we find acceptable the licensee's proposal to exclude the listed valves and associated penetrations from the Type C testing program, and the deletion of these valves and penetrations from Unit 1 TS Table 3.7-4.

2.3 Proposed Change 3 - Revise Unit 1 TS 4.7.A.2 and Unit 2 TS 4.6.1.3 to achieve similarity between the two documents, to comply with current 10 CFR Part 50, Appendix J testing requirements, and to specify an allowable leakage.

Appendix J, Section III.D.2.b requires that containment air locks be tested at 6-month intervals after initial fuel loading at an internal pressure of not less than Pa. Air locks that are opened when containment integrity is required shall be tested within 3 days after being opened, at a test pressure as specified in the TS. For Hatch Units 1 and 2, Pa is 57.5 psig and the test pressure specified for the 3-day test requirement is at least 10 psig. The existing Unit 1 TS provide no acceptance criteria for leakage resulting from the 3-day test, while Unit 2 states "no detectable seal leakage." Neither of these is suitable to meet the requirement of 10 CFR Part 50, Appendix J, Section III.D.2.b(iv), which requires that, "The acceptance criteria for air lock testing shall be stated in the Technical Specifications."

The change proposed by the licensee would amend Unit 1 TS 4.7.A.2 and Unit 2 TS 4.6.1.3 to achieve identical wording as regards the test requirements for the containment air locks, both in conformance with the requirements of

Appendix J, Sections III.D.2.b.(i), (ii), and (iii) and would state acceptance criteria for allowable leakage in accordance with Appendix J, Section III.D.2.b(iv) for the leak tests.

For the full pressure (Pa) leak tests, the allowable leakage is 0.05 La. This acceptance criterion is now stated in the TS for each unit and would remain unchanged. For the 3-day tests conducted at pressures $\leq$ 10 psig, an acceptable leakage for each set of door seals would be specified as 0.01 La. This reduced leakage rate is comparable to reduced pressure leak rates previously reviewed and approved by the staff for other plants, and is acceptable.

Accordingly, we find the licensee's proposed TS modifications regarding the containment air locks acceptable.

2.4 Proposed Change 4 - Delete penetration 218A from Unit 1 TS Table 3.7-2.

Amendment No. 140, issued on June 5, 1987, deleted penetration 218A from the listing of containment isolation valves subject to Appendix J leak rate testing (TS Table 3.7-4). Penetration 218A should have been deleted from TS Table 3.7-2 (Testable Penetrations with Double O-Ring Seals) at this same time. This proposed change would correct that oversight. The change is administrative in nature and is acceptable.

2.5 Proposed Change 5 - Delete the isolation valves associated with the primary feedwater and the torus drainage and purification systems from Unit 2 TS 3.6.1.2.

10 CFR Part 50, Appendix J, Section III.C states the conditions under which certain containment isolation valves sealed with fluid may be excluded when determining the total combined leakage rate.

Primary feedwater valves 2B21-F010A & B and 2B21-F077A & B (in penetrations 9A and 9B) are expected to remain covered by water following a design basis Loss-of-Coolant Accident (LOCA). Note 30 to Unit 2 FSAR Table 3.8-5 states that these valves are expected to remain covered by water following a design basis LOCA. Further, Note 8 to the same table states that the system remains filled with water post-LOCA, that the valves are tested with water, and that valve leakage is not included in the 0.60 La type B and C tests to determine total local leakage. These valves, therefore, should not have been listed in TS 3.6.1.2, and their removal from the listing merely amounts to an administrative correction.

Unit 2 FSAR subsection 6.2.1.2.2 states that the torus drainage and purification system valves (2G57-F011 and 2G51-F012), which may be open at the time of an accident, receive a signal to close, and that following closure a water seal is established by the suppression pool water. These valves thus will have no gaseous leakage and, in accordance with 10 CFR Part 50, Appendix J, Section III.C, they need not be considered in determining the combined leakage rate. The removal of these valves from the listing in TS 3.6.1.2 is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet 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 amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (41 FR 13765) on April 5, 1989, and consulted with the state of Georgia. No public comments were received, and the state of Georgia did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Lawrence P. Crocker, PDII-3/DRP-I/II

Dated: June 20, 1989

DATED June 20, 1989

AMENDMENT NO.101 TO FACILITY OPERATING LICENSE NPF-5, EDWIN I. HATCH, UNIT 2
AMENDMENT NO.164 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNIT 1

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