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

April 30, 1987

Posted
Amdt. 157
to DPR-47

Docket Nos.: 50-269, 50-270
and 50-287

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment Nos. 157, 157, and 154 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 - Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. 55689, 55690, 55691)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 157, 157, and 154 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated August 15, 1984, as revised on July 3, 1985.

The amendments revise TS 3.6-3 to reflect a new Limiting Condition for Operation (LCO) on the reactor building (RB) purge system. The RB purge system is required to be isolated whenever the reactor coolant system temperature is above 250°F and the pressure is above 350 psig. The LCO allows one isolation valve to open on each penetration at or below hot shutdown for testing or maintenance. TS 4.4.4 is added to reflect the RB purge system surveillance requirements and the purge valve seal inspection.

A copy of our Safety Evaluation is also included. Notice of issuance of the enclosed amendments will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Helen N. Pastis

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II

Enclosures:
1. Amendment No. 157 to DPR-38
2. Amendment No. 157 to DPR-47
3. Amendment No. 154 to DPR-55
4. Safety Evaluation

cc w/enclosures: See next page

Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated August 15, 1984, as revised July 3, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 157 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

B.J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: April 30, 1987

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated August 15, 1984, as revised July 3, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 157, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

B.J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: April 30, 1987

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated August 15, 1984, as revised on July 3, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

B.J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: April 30, 1987

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 157 TO DPR-38

AMENDMENT NO. 157 TO DPR-47

AMENDMENT NO. 154 TO DPR-55

DOCKET NOS. 50-269, 50-270, AND 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Page	Insert Page
iv	iv
3.6-1	3.6-1
3.6-2	3.6-2
3.6-3	3.6-3
	4.4-20

3.6 REACTOR BUILDING

Applicability

Applies to the containment when the reactor is in conditions other than refueling shutdown.

Objective

To assure containment integrity during shutdown (other than refueling shutdown), startup and operation.

Specification

3.6.1 Containment integrity shall be maintained whenever all three (3) of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater
b. Reactor coolant temperature is 200°F or greater
c. Nuclear fuel is in the core

3.6.2 Containment integrity shall be maintained whenever the reactor is subcritical by less than 1% $\Delta k/k$ or whenever positive reactivity insertions are being made which would result in the reactor being subcritical by less than 1% $\Delta k/k$.

3.6.3 Exceptions to 3.6.1 and 3.6.2 shall be as follows:

a. If either the personnel or emergency hatches become inoperable, except as a result of an inoperable door gasket, the hatch shall be restored to an operable status within 24 hours, or the reactor shall be in cold shutdown within the next 36 hours.

If a hatch is inoperable due to an inoperable door gasket:

1. The remaining door of the affected hatch shall be closed and sealed. If the inner door gasket is inoperable, momentary passage (not to exceed 10 minutes for each opening) is permitted through the outer door for repair or test of the inner door, provided that the outer door gasket is leak tested within 24 hours after opening of the outer door.

2. The hatch shall be restored to operable status within seven days or the reactor shall be in cold shutdown within the next 36 hours.

b. The Reactor Building purge supply and exhaust isolation valves shall be closed except as allowed by Specification 3.6.3.b.1 and 3.6.3.b.2.

1. The Reactor Building purge system may be operated, with the supply and exhaust isolation valves open, when the Reactor Coolant System temperature is below 250°F and pressure is below 350 psig.

2. For plant conditions when the Reactor Coolant System temperature is above 250°F and pressure is above 350 psig but the reactor is at or below hot shutdown, one Reactor Building Purge isolation valve on each penetration may be open for testing and/or maintenance per Specification 4.4.4.1 and 3.6.6.

3. For plant conditions other than contained in Specification 3.6.3.b.1, .2 above, with one or more Reactor Building purge valves open, the open valves shall be closed within one hour, or the plant shall be in hot shutdown within 12 hours and within an additional 24 hours, Reactor Coolant System temperature below 250°F and pressure below 350 psig.

c. A containment isolation valve, other than a Reactor Building Purge isolation valve, may be inoperable provided either:

1. The inoperable valve is restored to operable status within four hours.

2. The affected penetration is isolated within four hours by the use of a deactivated automatic valve secured and locked in the isolated position.

3. The affected penetration is isolated within four hours by the use of a closed manual valve or blind flange.

4. The reactor is in the hot shutdown condition within 12 hours and cold shutdown within 24 hours.

3.6.4 The reactor building internal pressure shall not exceed 1.5 psig or five inches of Hg if the reactor is critical.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and tagged.

3.6.6 The combined leakage rate for all penetrations and valves shall be determined in accordance with Specification 4.4.1.2. If, based on the most recent surveillance testing results the combined leakage rate exceeds the specified value and containment integrity is required then,

1) corrective action of Specification 3.6.3.c is met, or

2) repairs shall be initiated immediately and conformance with specified value shall be demonstrated within 48 hours or the reactor shall be in cold shutdown within an additional 36 hours.

Bases

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the Reactor Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here are conservative since an evaluation of National Weather Service records for this area indicates that from 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest of 30.85 inches of Hg.

Operation with a personnel or emergency hatch inoperable does not impair containment integrity since either door meets the design specifications for structural integrity and leak rate. Momentary passage through the outer door is necessary should the inner door gasket be inoperative to install or remove auxiliary restraint beams on the inner door to allow testing of the hatch. The time limits imposed permit completion of maintenance action and the performance of a local leak rate test when required or the orderly shutdown and cooldown of the reactor. Timely corrective action for an inoperable containment isolation valve is also specified.

When containment integrity is established, the limits of 10CFR100 will not be exceeded should the maximum hypothetical accident occur.

The Reactor Building purge system was designed to allow cleanup of the Reactor Building atmosphere. It is normally operated during a unit shutdown which will require entry into the Reactor Building. It is used to purge the Reactor Building with fresh air to reduce the contaminant levels within the Building atmosphere, thus reducing overall personnel exposure. At times, certain safety related functions necessitate entry into the Reactor Building prior to cold shutdown conditions. These include isolation of leaking primary coolant system valves and visual inspections following outages. Use of the purge system tends to minimize any personnel exposure while not significantly contributing to overall plant risk.

The Reactor Building Purge System is required to be isolated whenever the RCS temperature is above 250°F and pressure is above 350 psig. The maximum pressure limit of 350 psig is based on the Oconee Unit 1 NPSH curve for RC pump operation. This will give a reasonable operating margin for the pumps while operating the purge. The LCO allows one isolation valve to be open on each penetration at or below hot shutdown for testing/or maintenance.

REFERENCES

FSAR, Section 3.8

4.4.4 Reactor Building Purge System

Applicability

Applies to the Reactor Building Purge System.

Objective

To verify that the Reactor Building Purge System is operable.

Specification

4.4.4.1 Each shutdown, when the purge valves have been operated, leakage integrity tests shall be performed on the containment purge isolation valves after final closing and prior to going above hot shutdown. If the purge valves have not been operated, leakage integrity tests shall be performed prior to going above hot shutdown unless such tests have been conducted within the proceeding six months. If the acceptance criteria of Specification 4.4.1.2.3 are not met, Specification 3.6.6 shall apply. Unit shutdown to conduct the test and/or effect repairs is specifically not required.

4.4.4.2 Monthly, when the unit is above 250°F and 350 psig, the containment purge isolation valves shall be verified closed.

4.4.4.3 Each refueling the valve seals of the containment purge isolation valves shall be visually inspected and adjusted or replaced as appropriate.

4.4.4.4 Prior to use of the purge system at conditions between cold shutdown and 250°F and 350 psig, the isolation valves shall be exercise tested in accordance with the requirements (except test frequency) of the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

4.4.4.5 The pneumatically operated purge isolation valves shall be verified to close in response to a control signal from RIA-45 when the system is tested prior to refueling operations per Specification 3.8.10.

Bases

Leakage integrity tests of the purge supply and isolation valves are conducted in order to identify excessive degradation of the resilient seals. Excessive leakage past resilient seals is typically caused by severe environmental conditions and/or wear due to frequent use.

The pneumatically operated purge isolation valves are tested prior to refueling operations because the only automatic isolation system in service at refueling is through RIA-45, which only closes the pneumatic isolation valves.

UNITED STATES
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WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, and 3

DOCKET NOS. 50-269, 50-270 AND 50-287

I. INTRODUCTION

By application dated August 15, 1984 as revised on July 3, 1985, Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments would consist of changes to the Station's common TSs, TS 3.6.3, to reflect a new Limiting Condition for Operation (LCO) on reactor building (RB) purge system. The RB purge system is required to be isolated whenever th. reactor coolant system temperature is above 250°F and the pressure is above 350 psig. The LCO allows one isolation valve to be open on each penetration at or below hot shutdown for testing or maintenance. TS 4.4.4 is added to reflect the RB purge system surveillance requirements and the purge valve seal inspection.

II. BACKGROUND

By letter dated July 7, 1981, we requested the licensee to expand the TSs to cover the reactor building purge system isolation valves and the surveillance requirements for detecting seal deterioration in these valves. By letter dated May 10, 1983, the licensee responded to the staff request and indicated that such a technical specification was being developed. The licensee also provided technical justification for isolation of the reactor building purge system based on reactor coolant system temperature/pressure limits similar to those for the Low Pressure Injection System, (i.e. 350 psig and 250°F) rather than those normally specified for containment integrity (i.e. 300 psig and 200°F). By letter dated August 15, 1984, the licensee proposed Technical Specification 3.6.3 which consists of a limiting condition for operation requiring that the Reactor Building Purge System be isolated when the Reactor Coolant System (RCS) is above 250°F and 300 psig. However, the proposed technical specification would allow one isolation valve on each penetration to be open for testing and maintenance when the reactor coolant system is above the aforementioned limits but below hot shutdown conditions. Surveillance requirements on the purge valves were also provided. By letter dated July 3, 1985, the licensee proposed to increase the allowable RCS pressure limit to 350 psig.

III. EVALUATION

The licensee's proposal to incorporate slightly higher reactor coolant system pressure/temperature limits on the operation of the reactor building purge system isolation valves than normally specified for containment integrity (350 psig/250°F versus 300 psig/200°F) will permit the licensee to more expeditiously purge the Reactor Building during the period following shutdown, when purging is required to remove contaminants from the secondary side of the once-through steam generator (OTSG). This decontamination procedure requires that the OTSG be held at temperatures above 225°F with reactor coolant circulating for about 36 hours to minimize tube degradation during wet lay-up conditions. The proposed limits would save the licensee 27 to 30 critical path hours every cold shutdown. These limits are compatible with those for the Low Pressure Injection System and are not significantly different from those normally specified for containment integrity. Therefore, the licensee's proposed technical specification limits on purging present no adverse impact on safety.

The proposed surveillance requirements, TS 4.4.4, specify leakage testing after final closing of the purge valves before going above hot shutdown conditions unless such tests have been conducted within the preceeding six months. These surveillance requirements are in accordance with staff guidelines.

IV. SUMMARY

Based on the above evaluation, we conclude that the proposed TSs for the reactor building purge system isolation valves are in accordance with staff guidelines for assuring containment integrity. We find proposed TSs 3.6.3 and 4.4.4 to be acceptable.

V. ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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 these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these 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 these amendments.

VI. CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (50 FR 47860) on November 20, 1985, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 30, 1987

Principal Contributors: R. Ferguson
H. Pastis