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Amdt. 139
to DPR-47

MAY 30 1985

Dockets Nos. 50-269, 50-270
and 50-287

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Mr. Hal B. Tucker
Vice President - Nuclear Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 139, 139, and 136 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated February 13, 1984.

These amendments revise the TSs to update the TS reference to the Oconee Final Safety Analysis Report (FSAR) to ensure consistency with reference to the updated FSAR. Other changes requested in the February 13, 1984, submittal are still under NRC staff review and will be addressed by separate safety evaluation and license amendment.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance of the enclosed amendments will be included in the Commission's monthly notice.

Sincerely,

Original signed by

Helen Nicolaras, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 139 to DPR-38
2. Amendment No. 139 to DPR-47
3. Amendment No. 136 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	HNicolaras;cr	JStolz		GLainas
5/20/85	5/16/85	5/21/85	5/28/85	5/29/85

Duke Power Company

cc w/enclosure(s):

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated February 13, 1984, 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;

   D. The issuance of this 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 attachment 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. 139 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 30, 1985

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated February 13, 1984, 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;

   D. The issuance of this 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 attachment 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. 139 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 30, 1985

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.136
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated February 13, 1984, 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;

   D. The issuance of this 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 attachment 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. 136 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 30, 1985

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 139 TO DPR-38

AMENDMENT NO. 139 TO DPR-47

AMENDMENT NO. 136 TO DPR-55

DOCKETS 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 numbers and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
1-2	1-2
1-3	1-3
2.2-1	2.2-1
2.3-4	2.3-4
3.1-2	3.1-2
3.1-4	3.1-4
3.1-9	3.1-9
3.1-11	3.1-11
3.1-12	3.1-12*
3.1-13	3.1-13
3.1-16	3.1-16
3.1-18	3.1-18
3.2-2	3.2-2
3.3-6	3.3-6
3.6-3	3.6-3
3.13-1	3.13-1
3.16-1	3.16-1
4.3-1	4.3-1
4.4-5	4.4-5
4.4-6	4.4-6*
4.5-12	4.5-12
4.6-3	4.6-3
4.7-1	4.7-1
4.8-1	4.8-1
4.9-1	4.9-1
5.2-1	5.2-1*
5.2-2	5.2-2
5.3-1	5.3-1

*Overleaf page included for document completeness.

1.2.7 Refueling Operation

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 Startup

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.3 OPERABLE

A system, subsystem, train, component or device shall be considered OPERABLE when it is capable of performing its intended safety functions. Implicit in this definition shall be the assumption that all essential auxiliary equipment required in order to assure performance of the safety function is capable of performing its related support function(s). Auxiliary equipment includes but is not limited to normal or emergency electrical power sources, cooling and seal water, instrumentation and controls, etc. If either the normal or emergency power to system, subsystem, train, component or device is not available it is considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided:
(a) the alternate power source is available, and (b) the redundant system is operable.

1.4 PROTECTIVE INSTRUMENTATION LOGIC

1.4.1 Instrument Channel

An instrument channel is the combination of sensor, wires, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control and/or protection. An instrument channel may be either analog or digital in nature.

1.4.2 Reactor Protective System

The reactor protective system is shown in Figures 7.2-1 and 7.2-4 of the FSAR. It is that combination of protective channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protective channels, their associated instrument channel inputs, manual trip switch, all rod drive protective trip breakers and activating relays or coils.

1.4.3 Protective Channel

A protective channel as shown in Figure 7.2-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers and bistable modules provided for every reactor protective safety parameter) is a combination of instrument channels forming a single digital output to the protective system's coincidence logic. It includes a shutdown bypass circuit, a protective channel bypass circuit and reactor trip module and provision for insertion of a dummy bistable.

1.4.4 Reactor Protective System Logic

This system utilizes reactor trip module relays (coils and contacts) in all four of the protective channels as shown in Figure 7.2-1 of the FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one out of two times two logic. Each element of the one out of two times two logic is controlled by a separate set of two out of four logic contacts from the four reactor protective channels.

1.4.5 Engineered Safety Features System

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.3-1 of the FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant Engineered Safety Features equipment on a two-of-three basis for any given parameter.

1.4.6 Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

1.5 INSTRUMENTATION SURVEILLANCE

1.5.1 Trip Test

A trip test is a test of logic elements in a protective channel to verify their associated trip action.

1.5.2 Channel Test

A channel test is the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable.

1.5.3 Instrument Channel Check

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable.

1.5.4 Instrument Channel Calibration

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.

2.2 SAFETY LIMITS - REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1967.

Bases

The reactor coolant system [(1)] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110% of design pressure.[(2)] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under USAS Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established.[(3)] The settings, the reactor high pressure trip (2300 psig) and the pressurizer safety valves (2500 psig)[(4)] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the Reactor Coolant pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.

REFERENCES

(1) FSAR, Section 5

(2) FSAR, Section 5.2.3.10.1

(3) FSAR, Section 5.2.2.3, Table 5.4-7

(4) FSAR, Section 5.4.6, Table 5.1-1

Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1A, 2.3-1B, 2.3-1C. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip setpoint must be reduced to a value $\leq$5.0% of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip setpoint is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The over power trip setpoint of $\leq$5.0% prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0% of rated power if none of the reactor coolant pumps were operating.

Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped and is subject to the limitations set forth in Specification 3.1.8. The RPS trip setting limits and permissible instrument channels bypasses will be confirmed prior to single loop operation.

REFERENCES

(1) FSAR, Section 15.2.1

(2) FSAR, Section 15.7.1

(3) FSAR, Section 15.8.1

(4) FSAR, Sections 15.3.1, and 15.3.3

(5) FSAR, Section 15.6.3

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. (1) A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or low pressure injection pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One low pressure injection pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less. (2)

The low pressure injection system suction piping is designed for 300°F and 370 psig; thus the system with its redundant components can remove decay heat when the reactor coolant system is below this temperature. (3,4)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (5) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident at hot shutdown. (6) The pressurizer code safety valve lift setpoint shall be set at 2500 psig ±1% allowance for error and each valve shall be capable of relieving 300,000 lb/hr of saturated steam at a pressure no greater than 3% above the set pressure.

REFERENCES

(1) FSAR, Section 5.1.2.3

(2) FSAR, Section 6.3.3.2, and Tables 5.3-1, 5.4-2, 5.4-3, 5.4-6, 5.4-7, 5.4-8, and 6.3-2

(3) FSAR, Sections 5.4.7-1, and 9.3.3.2.3

(4) FSAR, Sections 5.4.7.4, and 6.3.3.2

(5) FSAR, Sections 5.2.3.10.4, and 5.4.6

(6) FSAR, Sections 5.2.3.7, and 15.2.3

Bases - Units 1, 2 and 3

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, startup and shutdown operations, and inservice leak and hydrostatic tests. The various categories of load cycles used for design purposes are provided in Table 5.2-1 of the FSAR.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1699 and BAW-1697.

The Figures specified in 3.1.2.1, 3.1.2.2 and 3.1.2.3 present the pressure-temperature limit curves for normal heatup, normal cooldown and hydrostatic tests respectively. The limit curves are applicable up to the indicated effective full power years of operation. These curves are adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations.

The cooldown limit curves are not applicable to conditions of off-normal operation (e.g., small LOCA and extended loss of feedwater) where cooling is achieved for extended periods of time by circulating water from the HPI through the core. If core cooling is restricted to meet the cooldown limits under other than normal operation, core integrity could be jeopardized.

The pressure-temperature limit lines shown on the figures specified in 3.1.2.1 for reactor criticality and on the figures referred to in 3.1.2.3 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region, or in test reactors.

The limitation on steam generator pressure and temperature provide protection against nonductile failure of the secondary side of the steam generator. At metal temperatures lower than the RT_{NDT} of +60°F, the protection against nonductile failure is achieved by limiting the secondary coolant pressure to 20 percent of the preoperational system hydrostatic test pressure. The

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a startup accident.[3]

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 4.3.2

(2) FSAR, Section 4.3.2.4

(3) FSAR, Section 15.3

appropriate by on site diffusion measurements using SF_6 (sulfur hexafluoride) as a gas tracer.

The combined gamma and beta whole body dose from a semi-infinite cloud is given by:

$$\text{Dose (Rem)} = 1/2[\bar{E}\cdot A\cdot V\cdot X/Q\cdot(3.7 \times 10^{10}\ \text{dps/Ci})\cdot(1.33 \times 10^{-11}\ \text{Rem/MeV/m}^3)]$$

$$\text{Dose (Rem)} = 0.246\cdot\bar{E}\cdot A\cdot V\cdot X/Q$$

$$A_{max}(\mu c/cc) = \frac{(\text{Dose})max}{0.246\cdot\bar{E}\cdot V\cdot X/Q} = \frac{0.5}{0.246 \times \bar{E} \times 78.25 \times 1.16 \times 10^{-4}}$$

$$A_{max}(\mu c/cc) = 224/\bar{E}$$

Where

A = Reactor coolant activity (μCi/ml = Ci/m^3)

V = Reactor coolant volume at 580°F leaked into secondary system (2763 ft^3 = 78.25 m^3)

X/Q = Atmospheric dispersion coefficient at site boundary for a two hour period (1.16×10^{-4} sec/m^3)

$\bar{E}$ = Average beta and gamma energies per disintegration (MeV) corrected to operating temperature and pressure.

Calculations required to determine $\bar{E}$ will consist of the following:

1. Quantitative measurement of the specific activity (in units of μCi/cc) of radionuclides with half lives longer than 30 minutes, which make up at least 95% of the total activity in reactor coolant samples.

2. A determination of the average beta and gamma decay energies per disintegration for each nuclide, measured in (1) above, by utilizing known decay energies and decay schemes (e.g., Table of Isotopes, Sixth Edition, March 1968).

3. A calculation of $\bar{E}$ by the average beta and gamma energy for each radionuclide in proportion to its specific activity, as measured in (1) above.

REFERENCE

FSAR, Section 15.9

3.1.5 Chemistry

Specification

3.1.5.1 If the concentration of oxygen in the primary coolant exceeds 0.1 ppm during power operation, corrective action shall be initiated within eight hours to return oxygen levels to ≤ 0.1 ppm.

3.1.5.2 If the concentration of chloride in the primary coolant exceeds 0.15 ppm during power operation, corrective action shall be initiated within eight hours to return chloride levels to ≤ 0.15 ppm.

3.1.5.3 If the concentration of fluorides in the primary coolant exceeds 0.15 ppm following modifications or repair to the primary system involving welding, corrective action shall be initiated within eight hours to return fluoride levels to ≤ 0.15 ppm.

3.1.5.4 If the concentration limits of oxygen, chloride or fluoride in 3.1.5.1, 3.1.5.2 and 3.1.5.3 above are not restored within 24 hours the reactor shall be placed in a hot shutdown condition within 12 hours thereafter. If the normal operational limits are not restored within an additional 24-hour period, the reactor shall be placed in a cold shutdown condition within 24-hours thereafter.

3.1.5.5 If the oxygen concentration and the chloride or fluoride concentration of the primary coolant system individually exceed 1.0 ppm, the reactor shall be immediately brought to the hot shutdown condition using normal shutdown procedure and action is to be taken immediately to return the system to within normal operation specifications. If normal operating specifications have not been reached in 12 hours, the reactor shall be brought to a cold shutdown condition using normal procedure.

Bases

By maintaining the chloride, fluoride and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack. (1,2)

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical and a residual of hydrazine is used when the reactor is subcritical to control the oxygen. The requirement that the oxygen concentration not exceed 0.1 ppm during power operation is added assurance that stress corrosion cracks will not occur. [4]

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchange resin, increase the hydrogen concentration in the makeup tank, etc.) and further because of the time dependent nature of any adverse effects arising from chlorides or oxygen concentrations in excess of the limits, it is unnecessary to shutdown immediately, since the condition can be corrected.

The oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time. (4) Thus, the period of eight hours to initiate corrective action and the period of 24 hours to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analyses are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the hot shutdown condition within 12 hours and corrective action will continue. If the operational limits are not restored within an additional 24 hour period, the reactor shall be placed in a cold shutdown condition within 24 hours thereafter.

The maximum limit of 1 ppm for the oxygen and halogen concentration that will not be exceeded was selected as the hot shutdown limit because these values have been shown to be safe at 500°F. (3)

<u>REFERENCES</u>

(1) FSAR, Section 5.2.1.7

(2) FSAR, Section 9.3.1-2

(3) Stress Corrosion of Metals, Logan

(4) Corrosion and Wear Handbook, O. J. DePaul, Editor

d. Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, coolant temperature, pressurizer water level and letdown storage tank level over a time interval. All of these indications are recorded. Since the pressurizer level is maintained essentially constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the letdown storage tank resulting in a tank level decrease. The letdown storage tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 1 inch of tank height. This inventory monitoring method is capable of detecting changes on the order of 31 gallons. A 1 gpm leak would therefore be detectable within approximately one half hour.

As described above, in addition to direct observation, the means of detecting reactor coolant leakage are based on 2 different principles, i.e., activity, sump level and reactor constant inventory measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the reactor building.

The upper limit of 30 gpm is based on the contingency of a complete loss of station power. A 30 gpm loss of water in conjunction with a complete loss of station power and subsequent cooldown of the reactor coolant system by the turbine bypass system (set at 1,040 psia) and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore electrical power to the station and makeup flow to the reactor coolant system.

REFERENCES

FSAR Sections 11.5.1, and 5.2.3.10.3

D. Dissolved boron concentration

This correction is for any difference in boron concentration, if required, between zero and full power. Since the moderator coefficient is more positive for greater dissolved boron concentrations, the sign of the correction depends on whether boron is added or removed. The correction is 0.16×10^{-6} $\Delta\alpha_m/\Delta PPM$. (The magnitude of the correction varies slightly with boron concentration; the value presented above, however, is valid for a range in boron concentrations from 1000 to 1400 ppm.)

E. Control rod insertion

This correction is for the difference in control rod worth (% $\Delta k/k$) in the core between zero and full power. The correction is 0.17×10^{-4} $\Delta\alpha_m/\%\Delta k/k$, where the sign for rod worth change is negative for rod insertion, because the moderator coefficient is more negative for a larger rod worth in the core.

F. Isothermal to distributed temperature

The correction for spatially distributed moderator temperature has been found to be less than or equal to zero. Therefore, zero is a conservative correction value for distributed effects.

G. Azimuthal xenon stability

Before commercial operation a test will be performed to verify that divergent azimuthal xenon oscillations do not occur.

REFERENCES

(1) FSAR, Section 15

(2) FSAR, Sections 4.2, and 4.3.2

(3) FSAR, Section 15.14

Bases

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank.(2)

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate the reactor coolant system to a 1% $\Delta k/k$ subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be re-evaluated with each reload. A minimum of 1020 ft^3 of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1835 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.7 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. The boric acid concentration of 8,700 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 77°F and therefore a temperature requirement of 87°F. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Sections 9.3.1, and 9.3.2

(2) FSAR, Figure 6.0.2

(3) Technical Specification 3.3

Three hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70°F without any control rods in the core. The minimum value specified in the tanks is 1835 ppm boron.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the Reactor Building design pressure will not be exceeded with one spray and two coolers operable. (4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable for seven days or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Unit 1, 2, and 3. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

Prior to initiating maintenance on any of the components, the redundant component(s) shall be tested to assure operability. Operability shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. The 24 hour period prior to removal is adequate to permit efficient scheduling of manpower and equipment testing while ensuring that the testing is performed directly prior to removal. The basis of acceptability is the low likelihood of failure within a clearly defined 48 hours following redundant component testing.

<u>REFERENCES</u>

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.3.3.2

(4) FSAR, Section 15.14.5

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

REFERENCE

FSAR, Section 3.8

3.13 SECONDARY SYSTEM ACTIVITY

Applicability

Applies to the limiting conditions of secondary system activity for operation of the reactor.

Objective

To limit the maximum secondary system activity.

Specification

The iodine-131 activity in the secondary side of a steam generator shall not exceed 1.4 μCi/cc.

Bases

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following a loss of load accident is considered. As stated in FSAR Section 15.8.2, 148,000 pounds of water is released to the atmosphere via the relief valves. A site boundary dose limit of 1.5 rem is used.

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser air ejector, thus, in the event of a loss of load incident there are only small quantities of these gases which would be released.

I-131 is the significant isotope because of its low MPC in air and because the other iodine isotopes have shorter half-lives, and therefore, cannot build up to significant concentrations in the secondary coolant, given the limitations on primary system leak rate and technical specification limiting activity. One-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plateout and retention in water droplets. I-131 is assumed to contribute 70% of the total thyroid dose based on the ratio of I-131 to the total iodine isotopes given in Table 11.2-2 of the FSAR.

The maximum inhalation dose at the site boundary is then as follows:

Dose (rem) = $C_i \cdot V \cdot B \cdot DCF \cdot (0.1) \cdot X/Q$
C = Secondary coolant activity (2.0 μCi/cc I-131 equivalent)
V = Secondary water volume released to atmosphere (90 m^3)
B = Breathing rate (3.47×10^{-4} m^3/sec)
X/Q = Ground level release dispersion factor (1.16×10^{-4} sec/m^3)
DCF = 1.48×10^6 rem/Ci
0.1 = Fraction of activity released

The resultant dose is 1.15 rem compared to the Radiation Protection Guide of 1.5 rem for an annual individual exposure in an unrestricted area.

3.16 HYDROGEN PURGE SYSTEM

Applicability

Applies to the Reactor Building Hydrogen Purge System.

Objective

To define the conditions necessary to assure the availability of the Reactor Building Hydrogen Purge System.

Specification

If the Reactor Building Hydrogen Purge System should become inoperable, it shall be restored to an operable status within 7 days or the Oconee Units shall be shutdown within 36 hours.

Bases

The hydrogen purge system is composed of a portable purging station and a portion of the Penetration Room Ventilation System. The purge system is operated as necessary to maintain the hydrogen concentration below the control limit. The purge discharge from the Reactor Building is taken from one of the Penetration Room Ventilation System penetrations and discharged to the unit vent. A suction may be taken on the Reactor Building via isolation valve PR-7 (Figure 6.0-5 of the FSAR) using the existing vent and pressurization connections.

The analysis to determine the effect on the incremental doses at the site boundary, resulting from purging hydrogen from the Reactor Building following a postulated LOCA, requires that the purge be started at 460 hours (19.2 days) following the LOCA to limit hydrogen concentration to 4% by volume. If the Hydrogen Purge System is determined to be inoperable, the requirement to restore the system to an operable status within seven days will provide reasonable assurance of its availability in the event of a LOCA.

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2200 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes. The specific code and edition thereof shall be consistent with 10 CFR 50.55.

REFERENCE

FSAR, Section 5

When containment integrity is established, the overall containment leak rate of 0.25 weight percent of containment air at 59 psig will assure that the limits of 10CFR100 will not be exceeded should the maximum hypothetical accident occur. In order to assure the integrity of the containment, periodic testing is performed at reduced pressure, 29.5 psig. The permissible leakage rate at this reduced pressure has been established from the initial integrated leak rate tests in conformance with 10CFR50, Appendix J.

The containment air locks (i.e., Personnel Hatch and Emergency Hatch) are tested on a more frequent basis than other penetrations. The air locks are utilized during periods of time when containment integrity is required as well as when the reactor is shutdown. Proper verification of door seal integrity is required to ensure containment integrity. Because the door seals are recessed, damage from tools due to air lock entry is improbable; however, a leak test of the outer door seals has been shown to be an acceptable alternative to the full hatch test to ensure air lock integrity.

REFERENCE

(1) FSAR, Sections 3.8.1.7.4, 6.2.4, and 14

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
1	Pressurizer liquid sample line (Unit 1 only)	Note 1	Type C	Note 2, 7b
2	OTSG A Sample line	Note 1	Type C	Note 7b
3	Component cooling inlet line	Note 1	Type C	Note 3, 7d
4	OTSG B drain line	Note 1	None required	Note 7b
5	RB normal sump drain line	Note 10	Type C	Note 7a, 7b, 9
6	Letdown line	Note 1	Type C	Note 2, 7b
7	RC Pump seal return line	Note 1	Type C	Note 3, 7b, 9
8	Loop A nozzle warming line	Not Vented	None required	Note 5, 7d
9	RCS normal makeup line and HP injection 'A' loop	Not Vented	None required	Note 5
10	RC Pump seal injection	Not Vented	Type C	Note 5, [illegible], 9, 12

4.5.4 Low Pressure Injection System Leakage

Applicability

Applies to Low Pressure Injection System leakage.

Objective

To maintain a preventive leakage rate for the Low Pressure Injection System which will prevent significant off-site exposures.

Specification

4.5.4.1 Acceptance Limit

The maximum allowable leakage from the Low Pressure Injection System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

4.5.4.2 Test

During each refueling outage, the following tests of the Low Pressure Injection System shall be conducted to determine leakage:

a. The portion of the Low Pressure Injection System, except as specified in (b), that is outside the containment shall be tested either by use in normal operation or by hydrostatically testing at 350 psig.

b. Piping from the containment emergency sump to the low pressure injection pump suction isolation valve shall be pressure tested at no less than 59 psig.

c. Visual inspection shall be made for excessive leakage from components of the system. Any excessive leakage shall be measured by collection and weighing or by another equivalent method.

Bases

The leakage rate limit for the Low Pressure Injection System is a judgment value based on assuring that the components can be expected to operate without mechanical failure for a period on the order of 200 days after a loss of coolant accident. The test pressure (350 psig) achieved either by normal system operation or by hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the Low Pressure Injection System (59 psig) is equivalent to the design pressure of the containment. The dose to the thyroid calculated as a result of this leakage is 0.76 rem for a two-hour exposure at the site boundary.

REFERENCE

FSAR, Sections 15.15.4, and 6.3.3.2.2

4.6.10 Annually, a one hour discharge service test at the required maximum load shall be made on the instrument and control batteries, the Keowee batteries, and the switching station batteries.

4.6.11 Monthly, the operability of the individual diode monitors in the Instrument and Control Power System shall be verified by imposing a simulated diode failure signal on the monitor.

4.6.12 Semiannually, the peak inverse voltage capability of each auctioneering diode in the 125 VDC Instrument and Control Power System shall be measured and recorded.

Bases

The Keowee Hydro units, in addition to serving as the emergency power sources for the Oconee Nuclear Station, are power generating sources for the Duke system requirements. As power generating units, they are operated frequently, normally on a daily basis at loads equal to or greater than required by Table 8.1-1 of the FSAR for ESF bus loads. Normal as well as emergency startup and operation of these units will be from the Oconee Unit 1 and 2 Control Room. The frequent starting and loading of these units to meet Duke system power requirements assures the continuous availability for emergency power for the Oconee auxiliaries and engineered safety features equipment. It will be verified that these units will carry the equipment of the maximum safeguards load within 25 seconds, including instrumentation lag, after a simulated requirement for engineered safety features. To further assure the reliability of these units as emergency power sources, they will be, as specified, tested for automatic start on a monthly basis from the Oconee control room. These tests will include verification that each unit can be synchronized to the 230 kV bus and that each unit can energize the 13.8 kV underground feeder.

The interval specified for testing of transfer to emergency power sources is based on maintaining maximum availability of redundant power sources.

Starting a Lee Station gas turbine, separation of the 100 kV line from the remainder of the system, and charging of the 4160 volt main feeder buses are specified to assure the continuity and operability of this equipment. The one hour time limit is considered the absolute maximum time limit that would be required to accomplish this.

REFERENCE

FSAR, Section 8

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Trip Insertion Time Test

Applicability

Applies to the surveillance of the control rod trip insertion time.

Objective

To assure the control rod trip insertion time is within that used in the safety analyses.

Specification

The control rod insertion time shall be measured at either full flow or no flow conditions as follows:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. For all rods at least once following each refueling outage.

The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant full flow conditions or 1.40 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR Chapter 15.

A rod is considered inoperable if the trip insertion time is greater than the specified allowable time.

REFERENCES

(1) FSAR, Section 15

(2) Technical Specification 3.5.2

4.8 MAIN STEAM STOP VALVES

Applicability

Applies to the main steam stop valves.

Objective

To verify the ability of the main steam stop valves to close upon signal and to verify the leak tightness of the main steam stop valves.

Specification

4.8.1 Using Channels A and B, the operation of each of the main steam stop valves shall be tested during each refueling outage to demonstrate a closure time of one second or less in Channel A and a closure time of 15 seconds or less for Channel B.

4.8.2 The leak rate through the main steam stop valves shall not exceed 25 cubic feet per hour at a pressure of 59 psig and shall be tested during each refueling outage.

Bases

The main steam stop valves limit the Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam line break accident. Their ability to promptly close upon redundant signals will be verified during each refueling outage. Channel A solenoid valves are designed to close all four turbine stop valves in 240 milliseconds. The backup Channel B solenoid valves are designed to close the turbine stop valves in approximately 12 seconds.

Using the maximum 15 second stop valve closing time, the fouled steam generator inventories and the minimum tripped rod worth with the maximum stuck rod worth, an analysis similar to that presented in FSAR Section 15.13 (but considering a blowdown of both steam generators) shows that the reactor will remain subcritical after reactor trip following a double-ended steam line break.

The main stop valves would become isolation valves in the unlikely event that there should be a rupture of a reactor coolant line concurrent with rupture of the steam generator feedwater header. The allowable leak rate of 25 cubic feet per hour is approximately 25 percent of total allowable containment leakage from all penetrations and isolation valves.

REFERENCES

(1) FSAR, Section 10.3.4, and 15.13

(2) Technical Specification 4.4.1

4.9 EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine-driven and motor-driven emergency feedwater pumps and associated valves.

Objective

To verify that the emergency feedwater pumps and associated valves are operable.

Specification

4.9.1 Pump Test

The turbine-driven and motor-driven feedwater pumps shall be operated on recirculation to the upper surge tank for a minimum of one hour in accordance with the requirements of Specification 4.0.4.

4.9.2 Valve Test

Automatic valves in the emergency feedwater flow path will be determined to be operable in accordance with the requirements of Specification 4.0.4.

4.9.3 System Flow Test

Prior to Unit operation above 25% Full Power following any modifications or repairs to the emergency feedwater system which could degrade the flow path and at least once per refueling cycle, the emergency feedwater system shall be given either a manual or an automatic initiation signal.

4.9.4 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly. In addition, during operation of the System Flow Test (Item 4.9.3 above), flow to the steam generators shall be verified by control room indication.

Bases

The monthly testing frequency is sufficient to verify that the emergency feedwater pumps are operable. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pumps. The parameters which are observed are detailed in the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI. The System Flow Test verifies correct total system operation following modifications or repairs.

REFERENCE

(1) FSAR, Section 10.4.7.4

5.2 CONTAINMENT

Specification

The containment for this unit consists of three systems which are the reactor building, reactor building isolation system, and penetration room ventilation system.

5.2.1 Reactor Building

The reactor building completely encloses the reactor and its associated reactor coolant system. It is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The structure can withstand the loss of 3 horizontal and 3 vertical tendons in the cylinder wall or adjacent tendons in the dome without loss of function. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal volume of the reactor building is approximately 1.91 x 10^6 cu. ft. The approximate inside dimensions are: diameter--116'; height--208 1/2'. The approximate thickness of the concrete forming the buildings are: cylindrical wall--3 3/4'; dome--3 1/4'; and the foundation slab--8 1/2'.

The concrete containment structure provides adequate biological shielding for both normal operation and accident situations. Design pressure and temperature are 59 psig and 286°F, respectively.

The reactor building is designed for an external atmospheric pressure of 3.0 psi greater than the internal pressure. This is greater than the differential pressure of 2.5 psig 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. Since the building is designed for this pressure differential, vacuum breakers are not required.

Penetration assemblies are seal welded to the reactor building liner. Access openings, electrical penetrations, and fuel transfer tube covers are equipped with double seals. Reactor building purge penetrations and reactor building atmosphere sampling penetrations are equipped with double valves having resilient seating surfaces. (1)

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 15 with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks.

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. (2)

5.2.3 Penetration Room Ventilation System

This system is designed to collect, control, and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slight negative pressure will be maintained in the penetration room to assure inleakage. (3)

REFERENCES

(1) FSAR Sections 6.2.1, and 6.2.3

(2) FSAR Section 6.2.3

(3) FSAR Section 6.5.1.1

5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies, all of which are prepressurized with Helium. (1)

5.3.1.2 The fuel assemblies shall form an essentially cylindrical lattice with an active height of 144 in. and an equivalent diameter of 128.9 in. (2)

5.3.1.3 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSR) distributed in the reactor core as shown in FSAR Figure 4.3-3. The full-length CRA and the APSR shall conform to the design described in the FSAR or reload report. (1)

5.3.1.4 Initial core and reload fuel assemblies and rods shall conform to design and evaluation described in FSAR or reload report and shall not exceed an enrichment of 3.5 percent of U-235.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the pressure components in the reactor coolant system shall be in accordance with the code requirements. (3)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F. (4)

5.3.2.3 The maximum reactor coolant system volume shall be 12,200 ft^3.

REFERENCES

(1) FSAR, Section 4.2.2

(2) FSAR, Section 4.3.1, and Table 4.3.1

(3) FSAR, Section 5.2.3.1

(4) FSAR Section 5.2.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

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

INTRODUCTION

By letter dated February 13, 1984, 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. Other changes requested in the February 13, 1984, submittal are still under staff review and will be addressed by separate safety evaluation and license amendment.

These amendments revise the TSs to update the TS references to the Oconee Final Safety Analysis Report (FSAR) to ensure consistency with reference to the updated FSAR.

EVALUATION

The initial Oconee FSAR update was provided, as required by 10 CFR Part 50.71, by letter dated July 19, 1982, from the licensee. The updated FSAR was revised in format to be consistent with present FSAR format criteria. This resulted in the FSAR references in the TSs being out of date. The updating of the references to the FSAR within the TSs assures that the appropriate sections of the FSAR are being identified. The updating of the TSs is an administrative change to achieve consistency with other documents.

During our review, we noted that several typographical errors appeared in the proposed amendment pages. We discussed these with the licensee, and he agreed to the changes. The correction of these typographical errors is strictly administrative in nature and does not affect the operation of the facility.

TS pages 1-4, 3.1-2, 3.1-11, 5.3-1 and 3.5-13 were withdrawn by the licensee and therefore, are no longer a part of the February 14, 1984 application.

ENVIRONMENTAL CONSIDERATION

These amendments relate to changes in administrative procedures or requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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.

CONCLUSION

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: May 30, 1985

Principal Contributor: H. Nicolaras