Duke Power Company
Oconee Nuclear Station
Attachment 1
Proposed Technical Specification Revision

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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.

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance on the secondary side of the steam generator considering all heat losses and additions.

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$\% \text{ Tilt} = 100 \left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right)$$

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in b below.

b. At least one door of the personnel hatch and the emergency hatch is closed and sealed during refueling or during personnel passage through these hatches.

c. All non-automatic containment isolation valves and blind flanges are closed as required.

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8.8 Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during Venting. Vent, used in system names, does not imply a venting process.

1.8.9 Member(s) Of The Public

Member(s) Of The Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

1.8.10 Unrestricted Area

An Unrestricted Area shall be any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial institutional and/or recreational purposes.

1.9 ACCESSIBLE/ACCESSIBILITY

A system, subsystem, train, component or device shall be accessible or have accessibility when it has been determined to be so by the group responsible for its surveillance or maintenance. The determination shall be based upon the environmental and radiological conditions at the system, subsystem, train, component or device location. The station safety supervisor or his designee may assist in providing the environmental evaluation and the station health physicist or his designee may assist in providing the radiological evaluation. The environmental evaluation shall be in accordance with standard industrial health and safety practices and regulations. The radiological evaluation shall be in accordance with the system health physics manual and ALARA principles.

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. Two conditions are imposed when
2.3-1B
2.3-1C
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 470 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

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 prevents 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

tionship between primary coolant pressure and temperature will be maintained relative ot the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

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:

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

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

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

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

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) corrected to operating temperature and pressure.

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).

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

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)

REFERENCES

(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

power α_m to the moderator coefficient at any power level above 15% power. For example, to correct to 100% power, α_m is adjusted by $(-.001 \times 10^{-4})$ (85%), which is $-.085 \times 10^{-4} \Delta\alpha_m$.

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 $0.16 \times 10^{-6}\ \Delta\alpha_m/\Delta PPM$. (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 $0.17 \times 10^{-4}\ \Delta\alpha_m/\%\Delta k/k$, 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 zenon stability

Before commercial operation a test will be performed to verify that divergent azimuthal xenon oscillaitons 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 accpetable 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.

REFERENCES

(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

REFERENCES

(1) FSAR, Section 4.3.2.3, Table 4.3-4

(2) FSAR, Section 15.12

(3) B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)
BAW-1396 (UNIT 2)
BAW-1400 (UNIT 3)

(4) Oconee 1, Cycle 4 - Reload Report - BAW 1447, March, 1977, Section 7.11

(5) Oconee 3, Cycle 6 - Reload Report - BAW-1634, August, 1980.

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 purposes 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 14.1.2.8.2, 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) = $Ci \cdot V \cdot B \cdot DCF \cdot (0.1) \cdot X/Q$
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 drom 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.

3.17.5 The fire hose stations listed in Table 3.17-1 shall be operable or the following action shall be taken:

1. If a fire hose station listed in Table 3.17.1 (except those in the Reactor Building which are inaccessible during power operation) is inoperable, an additional equivalent capacity fire hose of length sufficient to reach the unprotected area shall be provided at an operable hose station within 1 hour.

2. If the inoperable fire hose station cannot be restored to operable status within 14 days, continued operation of the affected unit is permitted provided that within the next 30 days a report is submitted to the Commission outlining the cause of the inoperability, actions taken, and the plans for restoring the system to operable status. Operation under this specification is not considered to be a degraded mode and is not reportable under Tech. Spec. 6.6.2.1.b(2).

3. Reactor Building fire hose stations listed in Table 3.17.1 shall be considered operable when water is available to isolation valves LPSW563 and LPSW564. In the event water is not available to these isolation valves, a minimum of 4 portable fire extinguishers shall be available outside containment in the Personnel Hatch area of the Auxiliary Building for fire brigade use upon entering the Reactor Building.

3.17.6 All fire barrier peentrations (including cable penetration barriers, fire doors, fire dampers) protecting safety related areas shall be operable.

If a fire barrier protecting a safety-related area is determined to be inoperable, the operability status of the fire detection instrumentation for the affected safety related area(s) shall be determined within 1 hour, and the following action shall be taken:

1. If the fire detection instrumentation for the affected area(s) is operable, a fire watch patrol shall be established to inspect the area at least once per hour.

2. If the fire detection instrumentation is inoperable, a continuous fire watch shall be established within the next hour on at least one side of the affected penetration fire barrier. The non-functional fire barrier penetration(s) shall be restored to functional status within 7 days.

3. If the non-functional fire barrier penetration(s) cannot be restored to functional status within 7 days, continued operation of the affected unit is permitted provided that within the next 30 days, a report is submitted to the Commission outlining the cause of the inoperability and the plans for restoring the system to operable status. Operation under this specification is not considered to be a degraded mode and is not reportable under Technical Specification 6.6.2.1.b(2).

Bases

Operability of the fire detection instrumentation ensures that adequate warnings capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to operability.

The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system spray and/or sprinklers, fire hose stations, and penetration fire barriers. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued operation of the nuclear plant.

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to a periodic inspections.

During periods of time when a barrier is not functional, a fire watch patrol will be required to inspect the affected area frequently as a precaution in addition to the fire detection instrumentation in the area. If fire detection instrumentation in the area is not operable, a continuous fire watch is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

TABLE 3.17-1 (cont'd)

C. Fire Hose Stations

Location No.	Valve No.	Area or Component Protected
3-J-28	2HPSW-241	1&2 3rd Floor Switchgear
3-M-43	3HPSW-339	3 3rd Floor Switchgear, 600V Load Center
AX-22	1HPSW-440	1 Battery Room
AX-20	2HPSW-440	2 Battery Room
AX-18	3HPSW-440	3 Battery Room
1RBH1	1LPSW-471	Ground Floor Level - East Side
2RBH1	2LPSW-471	Basement Floor Level - East Side
3RBH1	3LPSW-471	Basement - East Side
1RBH2	1LPSW-473	Intermediate Floor Level - East Side
2RBH2	2LPSW-473	Intermediate Floor Level - East Side
3RBH2	3LPSW-473	Intermediate Floor Level - East Side
1RBH3	1LPSW-475	Top of Shielding Floor Level - East Side
2RBH3	2LPSW-475	Top of Shielding Floor Level - East Side
3RBH3	3LPSW-475	Top of Shielding Floor Level - East Side
1RBH4	1LPSW-465	Top of Shielding Floor Level - West Side
2RBH4	2LPSW-465	Top of Shielding Floor Level - West Side
3RBH4	3LPSW-465	Top of Shielding Floor Level - West Side
1RBH5	1LPSW-467	Intermediate Floor Level - West Side
2RBH5	2LPSW-467	Intermediate Floor Level - West Side
3RBH4	3LPSW-467	Intermediate Floor Level - West Side
1RBH6	1LPSW-469	Ground Floor Level - West Side
2RBH6	2LPSW-469	Basement Floor Level - West Side
3RBH6	3LPSW-469	Basement - West Side

Basement - EL. 777'-6"
Ground - EL. 797'-6"
Intermediate - EL. 825'-0"
Top of Shielding - EL. 861'-0"

Keowee Hydro Station

Operating Deck (NW)	NA	Operating Floor
Operating Deck (NE)	NA	Operating Floor
Operating Deck (SW)	NA	Operating Floor
Operating Deck (SE)	NA	Operating Floor
Control Room	NA	Control Room
Mechanical Equipment Gallery	NA	Mechanical Equipment Gallery

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

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 signficant 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 judgement 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

Monthly, the turbine-driven and motor-driven feedwater pumps shall be operated on recirculation to the upper surge tank for a minimum of one hour.

4.9.2 Valve Test

Quarterly, automatic valves in the emergency feedwater flow path will be determined to be operable in accordance with the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

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

4.19 FIRE PROTECTION AND DETECTION SYSTEM

Applicability

This specification applies to fire protection and detection systems which protect systems and equipment required for safe shutdown.

Objective

To verify the operability of fire protection and detection systems.

Specification

4.19.1 Fire Detection Systems

a. Each of the fire detection instruments listed in Table 3.17-1 shall be tested for operability at least once per 6 months by performance of a Channel Functional Test, except as noted in part b.

b. The testing interval for detectors specified in Table 3.17.1 which are inaccessible during power operation may be extended until such time as the detectors become accessible for a minimum of 36 hours. The testing interval shall not extend past a refueling outage.

4.19.2 The Fire Suppression Water System shall be documented operable as follows:

a. Monthly

1. A functional test of the high pressure service water pump and associated automatic valve shall be performed.

2. Proper alignment of valves shall be verified.

3. A visual inspection of the fire hose stations listed in Table 3.17-1 (except those located in the Reactor Building which are inaccessible during power operations) shall be performed.

b. Annually

1. Each high pressure service water pump shall be tested to verify flow of 3000 gpm.

2. The sprinkler systems listed in Table 3.17-1 which protect safety-related systems shall be functionally tested, except in the cable spreading rooms, equipment rooms, and cable shafts.

3. The sprinkler system spray headers and nozzles, listed in Table 3.17.1, which protect safety-related systems, shall be inspected.

4. The fire hose stations (except those located in the Reactor Building which are inaccessible during power operations) shall receive a maintenance inspection to include removal and reracking of the hoses and inspection of coupling gaskets.

c. Refueling

1. A visual inspection of each nozzle's spray area will be conducted to verify the spray pattern is not obstructed.

2. Reactor Building fire hose stations which are inaccessible during power operation shall receive a maintenance inspection to include removal and reracking of the hoses and inspection of coupling gaskets.

d. At least once per 3 years:

1. A system flow test shall be performed on the fire suppression water system in accordance with Chapter 5, Section II of the Fire Protection Handbook, 14th Edition, NFPA.

2. The fire hose station valve listed in Table 3.17-1 shall be partial-stroke tested.

3. Each fire hose shall be subjected to a hydrostatic test at a pressure at least 50 psig greater than the maximum pressure at the station.

4.19.3 The high pressure CO_2 System for the generators at the Keowee Hydro Station shall be demonstrated operable as follows:

a. Monthly

1. Each valve in the flow path will be verified to be in its correct position.

b. Semiannually

1. The CO_2 storage tank weight shall be verified to be at least 90% of the full charge weight.

c. Refueling

1. The system shall be verified to actuate manually and automatically, upon receipt of a simulated action signal.

2. A flow test will be performed through headers and nozzles to assure no blockage.

4.19.4 Penetration fire barriers which protect safety-related equipment shall be verified functional by visual inspection at a refueling frequency and prior to declaring a penetration fire barrier functional following repairs or maintenance.

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 a nominal active height of 142 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

The spent fuel pool serving Units 1 and 2 is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of new and spent fuel assemblies predicted by the fuel management program.

Provisions are made in the Unit 1, 2 spent fuel pool to accommodate up to 1312 fuel assemblies and in the Unit 3 spent fuel pool up to 474 fuel assemblies.

5.4.2.2 Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

5.4.3 Except as provided in Specification 5.4.1.4, whenever there is fuel in the pool, the spent fuel pool is filled with water borated to the concentration that is used in the reactor cavity and fuel transfer canal during refueling operations.

5.4.4 The spent fuel pool and fuel transfer canal racks are designed for an earthquake force of 0.1g ground motion.

REFERENCE

FSAR, Section 9.1

Duke Power Company
Oconee Nuclear Station

Attachment 2

No Significant Hazards Consideration Evaluation

No Significant Hazards Consideration Evaluation

Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the Commission's regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). Example (i) of the types of amendments considered not likely to involve significant hazards considerations is applicable to this amendment request. This specific example involves amendment requests that are considered to be purely administrative changes to the Technical Specifications -- for example,

(a) a change to achieve consistency throughout the technical specification; or

(b) correction of an error; or

(c) a change in nomenclature.

The proposed Technical Specifications amendment addressed in this submittal has been determined by Duke to be an administrative change. The specific requested changes are required so that the technical specification will be consistent thoughout and past omissions will be corrected.

Briefly, the proposed amendment defines the term accessible/accessibility; incorporates the fire hose stations located in the three Oconee reactor buildings into the limited conditions of operation (LCO) and surveillance requirements addressing the fire protection and detection systems; and also, updates the Technical Specification references to the Oconee Nuclear Station Final Safety Analysis Report (FSAR).

Duke has determined, based on the consideration that the requested amendment is administrative in nature, that the revisions do not involve a significant increase in the probability or consequences of accidents previously considered, nor create the possibility of a new or different kind of accident, and will not involve a significant decrease in a safety margin. Therefore, Duke concludes that there is no significant hazards consideration involved in this amendment request.

Attachment 3

Technical Justification

The proposed Technical Specification revisions addressed in this submittal are administrative in nature and, as such, are of no public health or safety significance. Briefly, the proposed amendment defines the term accessible/accessibility; updates the Technical Specifications references to the Oconee Nuclear Station Final Safety Analysis Report (FSAR); and, also, incorporates the fire hose stations located in the Reactor Building into the Technical Specifications.

The definition of the term accessible/accessibility, as proposed in this amendment, will provide for uniform interpretation of this term throughout the Technical Specifications. This will assure a consistent application of this term.

The initial Oconee FSAR update was provided as required by 10 CFR 50,§50.71 by a Duke letter dated July 19, 1982. The updated FSAR was reformatted to be consistent with present FSAR format criteria. This resulted in the FSAR references within the Technical Specifications being out of date. The updating of the references to the FSAR within the Technical Specifications assures that the appropriate sections of the FSAR are being identified. The updating of the Technical Specifications is to be an administrative change to achieve consistency with other documents.

By letters dated March 18 and May 15, 1981, Duke proposed an amendment to the Oconee Technical Specifications relating to Oconee's fire protection program. Approval of the amendment request was issued by a June 9, 1981 letter. Recently, it was determined that the Section 3.17 and Section 4.19 (Fire Protection and Detection System) do not address the fire hose stations located in each of the three Reactor Buildings. This is considered to be an error of omission. The change included in this proposed amendment corrects this error. This change is considered to be purely administrative in that it corrects the error of omission from the Technical Specifications. The technical acceptability of the fire hose locations has been previously reviewed by the NRC and has been documented in several inspection reports (Inspection Report Nos. 50-269/80-7, 50-270/80-5 and 50-287/80-5; Inspection Report Nos. 50-269/80-14, 50-270/80-9 and 50-287/80-9; Inspection Report Nos. 50-269/270/287/81-12). Through these reports, inspectors have reviewed the fire protection program at Oconee and have not identified any deficiencies relative to the locations of fire hose stations in the Oconee Reactor Buildings.