OCONEE NUCLEAR STATION

Proposed Technical Specification Revision

Pages
3.4-1
3.4-2
3.5-2
3.5-2A
3.5-5
4.1-9
4.9-1

Section		Page
3.4	SECONDARY SYSTEM DECAY HEAT REMOVAL	3.4-1
3.5	INSTRUMENTATION SYSTEMS	3.5-1
3.5.1	Operational Safety Instrumentation	3.5-1
3.5.2	Control Rod Group and Power Distribution Limits	3.5-6
3.5.3	Engineered Safety Features Protective System Actuation Setpoints	3.5-25
3.5.4	Incore Instrumentation	3.5-27
3.6	REACTOR BUILDING	3.6-1
3.7	AUXILIARY ELECTRICAL SYSTEMS	3.7-1
3.8	FUEL LOADING AND REFUELING	3.8-1
3.9	RELEASE OF LIQUID RADIOACTIVE WASTE	3.9-1
3.10	RELEASE OF GASEOUS RADIOACTIVE WASTE	3.10-1
3.11	MAXIMUM POWER RESTRICTIONS	3.11-1
3.12	REACTOR BUILDING POLAR CRANE AND AUXILIARY HOIST	3.12-1
3.13	SECONDARY SYSTEM ACTIVITY	3.13-1
3.14	SHOCK SUPPRESSORS (SNUBBERS)	3.14-1
3.15	PENETRATION ROOM VENTILATION SYSTEMS	3.15-1
3.16	HYDROGEN PURGE SYSTEM	3.16-1
3.17	FIRE PROTECTION AND DETECTION SYSTEMS	3.17-1
4	SURVEILLANCE REQUIREMENTS	4.0-1
4.0	SURVEILLANCE STANDARDS	4.0-1
4.1	OPERATIONAL SAFETY REVIEW	4.1-1
4.2	REACTOR COOLANT SYSTEM SURVEILLANCE	4.2-1
4.3	TESTING FOLLOWING OPENING OF SYSTEM	4.3-1
4.4	REACTOR BUILDING	4.4-1

3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

3.4.1 Emergency Feedwater System

The reactor shall not be heated above 250°F unless the following conditions are met:

a. Three emergency feedwater pumps and associated flow paths shall be operable with one steam-driven pump capable of being powered from an operable steam supply system and two motor-driven pumps.

b. If one emergency feedwater pump is inoperable then, restore it to operable status within 60 hours. Otherwise, the unit shall be in a hot shut down condition within an additional 12 hours and below 250°F in another 12 hours.

c. If no 100% flow path is operable, then the unit shall be subcritical within 1 hour, and brought to cold shutdown within an additional 12 hours or at the maximum safe shutdown rate.

3.4.2 The 16 steam system safety valves shall be operable.

3.4.3 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hotwell.

3.4.4 The emergency condenser circulating water system shall be operable.

3.4.5 The controls of the emergency feedwater system shall be independent of the Integrated Control System.

Bases

The Main Feedwater System and the Turbine Bypass System are normally used for decay heat removal and cooldown above 250°F. Feedwater makeup is supplied by operation of a hotwell pump, condensate booster pump and a main feedwater pump.

The Emergency Feedwater (EFW) System assures sufficient feedwater supply to the steam generators of each unit, in the event of loss of the main Feedwater System, to remove energy stored in the core and primary coolant. The EFW System is designed to provide sufficient secondary side steam generator heat sink to enable cooldown from reactor trip power operation down to cold shutdown conditions.

A 100% emergency feedwater flowpath shall be considered to be either: 1) the steam driven turbine pump, associated valves and piping capable of feeding either steam generator or 2) both motor driven pumps, associated valves and piping each capable of feeding the associated steam generator.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps or low main feedwater header pressure. The EFW System will supply sufficient feedwater for approximately five-hour cooldown at a flowrate of at least 720 gpm to enable the Reactor Coolant System to reach conditions at which the Decay Heat Removal System may be operated.

Two motor-driven emergency feedwater pumps are installed in each unit in addition to the steam-driven emergency feedwater pump. The motor-driven pumps are powered from diverse emergency power supplies.

All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Normally, decay heat is removed by steam relief through the turbine bypass system to the condenser. Condenser cooling water flow is provided by a siphon effect from Lake Keowee through the condenser for final heat rejection to the Keowee Hydro Plant tailrace. Decay heat can also be removed from the steam generators by steam relief through the main steam relief valves.

The minimum amount of water in the upper surge tank, condensate storage tank and hotwell is the amount needed for 11 hours of operation per unit. This is based on the conservative estimate of normal makeup being 0.5% of throttle flow. Throttle flow at full load, 11,200,000 lbs/hr., was used to calculate the operation time. For decay heat removal the operation time with the volume of water specified would be considerably increased due to the reduced throttle flow.

The total relief capacity of the 16 steam system safety valves is 13,105,000 lbs/hr.

REFERENCE

FSAR, Section 10

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and two channels each of the following are operable: four reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels, four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and high reactor building pressure instrument channels. The engineered safety features actuation system must have two analog channels functioning correctly prior to a startup.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column B (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE-279 as described in FSAR Section 7.

There are four reactor protective channels. A fifth channel that is isolated from the reactor protective system is provided as a part of the reactor control system. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other channels is one out of two.

The four reactor protective channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protective system bypass switch key permitted in the control room. That key will be under the administrative control of the Shift Supervisor. Spare keys will be maintained in a locked storage accessible only to the station Manager.

Each reactor protective channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. There are four shutdown bypass keys in the control room under the administrative control of the Shift Supervisor. The use of a key operated shutdown bypass switch for on-line testing or maintenance during reactor power operation has no significance when used in conjunction with a key operated channel bypass switch since the channel trip relay is locked in the untripped state. The use of a key operated shutdown bypass switch alone during power operation will cause the channel to trip. When the shutdown bypass switch is operated for on-line testing or maintenance during reactor power operation, reactor power and RCS pressure limits as specified in Table 2.3-1A, B, or C are not applicable.

The source range and intermediate range nuclear instrumentation overlap by one decade of neutron flux. This decade overlap will be achieved at 10^{-10} amps on the intermediate range instrument.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 600 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state on-line repairs to the failed device, when practical, will be made, and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

The loss of main turbine and loss of feedwater anticipatory reactor trips are required only above 20% FP. For power levels less than 25% FP, sufficient time for operator action exists to initiate feedwater, and for the turbine trip event, the system has sufficient responsiveness such that a reactor trip or turbine trip is not anticipatory. The value of 20% FP has been selected to provide operational flexibility during startup and shutdown of the unit.

REFERENCE

FSAR, Section 7.1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (Cont'd)

	Functional Unit	(A) Minimum Operable Analog Channels	(B) Minimum Degree of Redundancy	Operator Action If Conditions Of Column A and B Cannot Be Met
	b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (3)
15.	Turbine Stop Valves Closure	2	1	Bring to hot shutdown within 12 hours (f)
16.	Loss of Main Turbine, Loss of Feedwater Reactor Trip	1	0	Bring to hot shutdown within 12 hours. (g)

(a) For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

(d) Single loop operation at power (after testing and approval by the NRC) is not permitted unless the operating channels are the two receiving Reactor Coolant Temperature from operating loop.

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in the cold shutdown condition within 24 hours.

(f) One operable channel with zero minimum degree of redundancy is allowed for 24 hours before going to the hot shutdown condition.

(g) Required only when reactor is greater than 20% reactor power.

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rod Movement (1)	Movement of Each Rod	Monthly
2.	Pressurizer Safety Valves	Setpoint	50% Annually
3.	Main Steam Safety Valves	Setpoint	25% Annually
4.	Refueling System Interlocks	Functional	Prior to Refueling
5.	Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System Leakage (2)	Evaluate	Daily
7.	Condenser Cooling Water System Gravity Flow Test	Functional	Annually
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9.	Spent Fuel Cooling System	Functional	Prior to Refueling
10.	High Pressure and Low Pressure Injection System (3)	Vent Pump Casings	Monthly and Prior
11.	Control Grade Reactor Trip Loss of Main Turbine, Loss of Feedwater	Functional	Monthly
12.	Emergency Feedwater Pump Automatic Start Feature	Functional	18 months

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

4.9 EMERGENCY FEEDWATER PUMP PERIODIC TESTING

Applicability

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

Objective

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

Specification

4.9.1 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 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.

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.

REFERENCES

(1) FSAR, Section 10.2.2

(2) FSAR, Section 14.1.2.8.3