U.S. Nuclear Regulatory Commission
LIC-93-0159

ATTACHMENT A

TABLE OF CONTENTS (Continued)

DEFINITIONS

PROTECTIVE SYSTEMS (Continued)

Engineered Safety Feature Logic(2)

The system which utilizes relay contact outputs from individual instrument channels to provide a dual channel signal to independently initiate the actuation of the engineered safety feature equipment. Two logic subsystems, termed A and B, are provided; each subsystem is composed of four channels wired to provide independent safety feature initiation signals on a 2-out-of-4 basis.

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.

INSTRUMENTATION SURVEILLANCE

Channel Check

A qualitative determination of acceptable operability by observation of channel behavior ~~behaviour~~ during normal plant operation. This determination shall where feasible, include comparison of the channel with other independent channels measuring the same variable.

Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating action.

Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarms, interlocks or trip, and shall be deemed to include the channel functional test.

Source Check

Verification of channel response when the channel sensor is exposed to a radioactive source.

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.2 Safety Limit, 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 to the containment.

Specification

The reactor coolant system pressure shall not exceed 2750 psia when fuel assemblies are located within the reactor vessel.

Basis

The reactor coolant system serves as a barrier to prevent radionuclides in the reactor coolant from reaching the containment atmosphere. (1) In the event of a fuel cladding failure, the reactor coolant system is the primary barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of the reactor coolant system and fuel cladding. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS Section B31.1 is 110% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established. (2)

The settings and capacity of the main steam ~~steam system~~ safety valves (1000-1050 psia)(3), the reactor high-pressure trip (≤ 2400 psia) and the reactor coolant system safety valves (2500-2545 psia)(4) have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test pressure was conducted at 3125 psia (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves consistent with the reactor high pressure trip, ~~at 2400 psia~~ and opening the steam system steam dump and bypass valves upon receipt of a turbine trip signal.(5)

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.3 Limiting Safety System Settings, Reactor Protective System (continued)

(3) High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the reactor and steam system safety valves to prevent reactor coolant system overpressure (Specification 2.1.6). In the event of loss of load without reactor trip, the temperature and pressure of the reactor coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is ~~100 psi~~ below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.[1]

(4) Thermal Margin/Low Pressure Trip - The thermal margin/low pressure trip is provided to prevent operation when the DNBR is less than 1.18, including allowance for measurement error. The thermal and hydraulic limits shown in the Thermal Margin/Low Pressure 4 Pump Operation Figure, contained in the COLR, define the limiting values of reactor coolant pressure, reactor inlet temperature, axial shape index, and reactor power level which ensure that the thermal criteria[8] are not exceeded. The low set point of a 1750 psia trips the reactor in the unlikely event of a loss-of-coolant accident. The thermal margin/low pressure trip set points shall be set according to the equation given in the COLR for the Thermal Margin/Low Pressure Limit.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (continued)

2.1.6 Pressurizer and Main Steam Safety Valves (continued)

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to operable status or close the block valve(s). Otherwise, be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Basis

The highest reactor coolant system pressure reached in any of the accidents analyzed resulted from a complete loss of turbine generator load without simultaneous reactor trip while operating at 1500 MWt.(2) This pressure was less than the 2750 psia safety limit and the ASME Section III upset pressure limit of 10% greater than the design pressure.(1) The reactor is assumed to trip on a "High Pressurizer Pressure" trip signal. [INSERT 1]

The power-operated relief valves (PORV's) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount ~~or~~ of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

INSERT 1

The pressurizer safety valves are required to be calibrated to within ± 1% of the specified setpoint value using ASME Section XI test methods. ASME Section XI requires that valves in steam service use steam as the test medium for establishing the setpoint. With the presence of a water-filled loop seal, establishing the valve setpoint with steam may result in in-situ valve actuation at pressures outside the ± 1% tolerance specified. Under transient conditions, it is expected that the valve(s) will actuate at no less than 4% below, nor greater than 6% above, the specified setpoint, which is within the tolerance assumed in the safety analysis.[2]

and LCV-218-3

d1. The required BAST volume of Figure 2-11 can be combined between CH-11A and CH-11B when both tanks are operable.

d2. When LCV-218-3 is inoperable or the SIRW tank volume is below Technical Specification 2.2(1) minimum, then each BAST must be operable and contain the required volume of Figure 2-11 corresponding to the requirements of the SIRW tank Technical Specification boron concentration.

d3. When BAST CH-11B is inoperable, then BAST CH-11A must be operable and contain the required volume of Figure 2-11 and LCV-218-3 must be operable.

d4. When BAST CH-11A is inoperable, then BAST CH-11B must be operable and contain the required volume of Figure 2-11 and LCV-218-3 must be operable.

e. Level instruments on the inservice BAST shall be operable.

(3) Modification of Minimum Requirements

During power operation, the minimum requirements may be modified to allow any one of the following conditions to exist at any one time. If the system is not restored to meet the minimum requirements within the time period specified, the reactor shall be placed in the hot shutdown condition in 4 hours and in the cold shutdown condition within an additional 48 hours.

a. One of the operable charging pumps may be removed from service provided two charging pumps are operable within 24 hours.

b. Both boric acid pumps may be out of service for 24 hours provided that both BASTs meet the requirements of Figure 2-11.

c. One level instrument channel on each inservice concentrated boric acid tank may be out of service for 24 hours.

d. One BAST may be removed from service for 72 hours provided that either of the conditions of 2.2(2)d3 or 2.2(2)d4 above is met.

Basis

The chemical and volume control system provides control of the reactor coolant system boron inventory.[1] This is normally accomplished by using any one of the three charging pumps in series with one of the two boric acid pumps. An alternate method of boration will be to use the charging pumps directly from the SIRW storage tank. A third method will be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three different paths.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System

Applicability

Applies to the operating status of the emergency core cooling system.

Objective

To assure operability of equipment required to remove decay heat from the core.

Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least the refueling boron concentration at a temperature not less than 50°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig with a tank ~~liquid~~ level of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection pump is operable on each ~~bus.~~ associated 4,160 V engineered safety feature bus.

f. One high-pressure safety injection pump is operable on each ~~bus.~~ associated 4,160 V engineered safety feature bus.

g. Both shutdown heat exchangers and three of four component cooling heat exchangers are operable.

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from the motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the reactor vessel head, a pressurizer safety valve, or a PORV is removed.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 312°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 271°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable, a single HPSI pump may be made operable and utilized for boric acid injection to the core.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing CEA's and diluting boron in the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. ~~During low power physics tests at low temperatures, there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards systems are not required.~~

The SIRW tank contains a minimum of 283,000 gallons of usable water containing a boron concentration of at least the refueling boron concentration. This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 60°F.[2]

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3. Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.10 Reactor Core (Continued)

2.10.4 Power Distribution Limits (Continued)

In order for these objectives to be met, the reactor must be operated consistent with the operating limits specified for margin to DNB.

The parameter limits given in (5) and the F_R^T, F_{xy}^T and Core Power Limitations Figure provided in the COLR along with the parameter limits on quadrant tilt and control element assembly position (Power Dependent Insertion Limit Figure provided in the COLR) provide a high degree of assurance that the DNB overpower margin will be maintained during steady state operation.

The actions specified assure that the reactor is brought to a safe condition.

The reactor coolant pump differential pressure monitoring system may ~~that will~~ be used to measure flow. ~~provides an accurate method of determining reactor coolant flow.~~

~~The procedure for determining individual pump and reactor vessel flow will be as follows:~~

~~1. Obtain a pump causing ΔP, using the precision resistor and high accuracy digital voltmeter and converting to pressure.~~

~~2. Obtain cold leg temperature and pressurizer pressure.~~

~~3. Correct the reading to the curve specific gravity.~~

~~4. Obtain pump flows from individual pump casing vs. flow curves.~~

~~5. Add the individual pump flows to obtain the best estimate reactor vessel flow.~~

2.0 LIMITING CONDITIONS FOR OPERATION

2.11 Containment Building and Fuel Storage Building Crane

Applicability

Applies to the use of cranes over the reactor coolant system and the spent fuel storage pool.

Objective

To specify restrictions on the use of the overhead cranes in the Containment Building and the Auxiliary Building.

Specifications

Use of the Containment Building and the Auxiliary Building overhead cranes shall be subject to the following limiting conditions.

(1) The Containment polar crane shall not be used to transport loads over the reactor coolant system if the temperature of the coolant or steam in the pressurizer exceeds 225°F.

(2) The Auxiliary Building crane shall not be used to move material over irradiated fuel in the fuel storage pool. If the crane interlocks are inoperable or bypassed, the crane operation will be under the direct control of a supervisor.

Basis

Loads are not to be allowed over the pressurized reactor coolant system to preclude dropping objects which could rupture the boundary of the reactor coolant system allowing loss of coolant and over-heating of the core.

The Auxiliary Building crane is provided with an electrical interlock system that will normally prevent the trolley from moving over the storage pool.(1) This minimizes the possibility of dropping an object on the irradiated fuel stored in the pool and resulting in the release of radioactive products. The interlocks may be bypassed under strict administrative control to allow required movement of fuel and material over the pool. The crane can be used over the equipment hatches and areas located in the north and west ends of the Auxiliary Building and over the railroad siding without the interlocks operable since a load, even if dropped, could not fall into the storage pool.

References

(1) ~~FSAR~~ USAR, Section 14.18

(3) Containment High Radiation (Air Monitoring) (Continued)

~~The setpoints for the isolation function will be calculated in accordance with the ODCM.~~

(4) Low Steam Generator Pressure

A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a ± 22 psi uncertainty and was the setting used in the safety analysis.[3]

Closure of the MSIVs (and the bypass valves, along with main feedwater isolation and bypass valves) is accomplished by the steam generator isolation signal which is a logical combination of low steam generator pressure or high containment pressure.

As part of the AFW actuation logic, a separate signal is provided to terminate flow to a steam generator upon sensing a low pressure in that steam generator if the other steam generator pressure is greater than the pressure setting. This is done to minimize the temperature reduction in the reactor coolant system in the event of a main steam-line break. The setting of 466.7 psia includes a +31.7 psi uncertainty; therefore, a setting of 435 psia was used in the safety analysis.

(5) SIRW Tank Low Level

Level switches are provided on the SIRW tank to actuate the valves in the safety injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of approximately 24 minutes following a safety injection signal. The switch-over point of 16 inches ~~above~~ tank bottom is set to prevent the pumps from running dry during the ~~10 seconds~~ time required to stroke the valves and to hold in reserve approximately 28,000 gallons of water of at least the refueling boron concentration. The ~~FSAR~~ USAR loss of coolant accident analysis[4] assumed the recirculation started when the minimum usable volume of 283,000 gallons had been pumped from the tank.

Effluent radiation monitor isolation function setpoints will be calculated in accordance with the ODCM. Process radiation monitor setpoints will be calculated in accordance with the applicable Chemistry Manual calibration procedure.

TABLE 2-1

Engineered Safety Features System Initiation Instrument Setting Limits

	Functional Unit	Channel	Setting Limit
1.	High Containment Pressure	a. Safety Injection (3) b. Containment Spray c. Containment Isolation d. Containment Air Cooler DBA Mode e. Steam Generator Isolation	≤ 5 psig
2.	Pressurizer Low/Low Pressure	a. Safety Injection (3) b. Containment Spray c. Containment Isolation d. Containment Air Cooler DBA Mode	≥ 1600 psia (1)
3.	Containment High Radiation	Containment Ventilation Isolation	In accordance with the ~~Offsite Dose Calculational Manual~~ applicable Chemistry Manual calibration procedure (2)
4.	Low Steam Generator Pressure	a. Steam Line Isolation b. Auxiliary Feedwater Actuation	≥ 500 psia ≥ 466.7 psia
5.	SIRW Low Level Switches	Recirculation Actuation	16 inches +0, -2 in. above tank bottom
6.	4.16 KV Emergency Bus Low Voltage	a. Loss of Voltage	(2995.2 ± 104) volts ≤ 5.9 (4) $^{20.9}$ seconds Trip
		b. Degraded Voltage i) Bus 1A3 Side	≥ 3825.52 volts (4.8 ± .5) seconds Trip

ventilation isolation signals available if the containment ventilation isolation valves are closed. If after 24 hours from time of initiating a hot shutdown procedure the inoperable engineered safety features or isolation functions channel has not been ~~stored~~ restored to operable status, the reactor shall be placed in a cold shutdown condition within the following 24 hours. This specification applies to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(3) In the event the number of channels of a particular system in service falls below the limits given in the columns entitled "Minimum Operable Channels" ~~of~~ or "Minimum Degree of Redundancy", except as conditioned by the column entitled "Permissible Bypass Conditions", the reactor shall be placed in a hot shutdown condition within 12 hours; however, operation can continue without containment ventilation isolation signals available if the ventilation isolation valves are closed. If minimum conditions for engineered safety features or isolation functions are not met within 24 hours from time of discovering loss of operability, the reactor shall be placed in a cold shutdown condition within the following 24 hours. If the number of operable high rate trip-wide range log channels falls below that given in the column entitled "Minimum Operable Channels" in Table 2-2 and the reactor is at or above 10^{-4}% power and at or below 15% of rated power, reactor critical operation shall be discontinued and the plant placed in an operational mode allowing repair of the inoperable channels before startup or reactor critical operation may proceed.

If, during power operation, the rod block function of the secondary CEA position indication system and rod block circuit are inoperable for more than 24 hours, or the plant computer PDIL alarm, CEA group deviation alarm and the CEA sequencing function are inoperable for more than 48 hours, the CEAs shall be withdrawn and maintained at fully withdrawn and the control rod drive system mode switch shall be maintained in the off position except when manual motion of CEA Group 4 is required to control axial power distribution.

(4) In the event that any of the following Alternate Shutdown Panel instrumentation or control circuits become inoperable, either restore the inoperable component(s) to operable status within seven days, or be in hot shutdown within the next twelve hours. This specification is applicable in Modes 1 and 2.

Wide Range Logarithmic Power (AI-212)

Source Range Power (AI-212)

Reactor Coolant Cold Leg Temperature (AI-185)

Reactor Coolant Hot Leg Temperature (AI-185)

Pressurizer Level (AI-185)

Volume Control Tank Level (AI-185)

TABLE 2-3
(Continued)

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
5	Auxiliary Feedwater				
A	Manual	1	None	None	N/A
B	Auto. Initiation A B			Operating Modes 3, 4, and 5	
	-Steam Generator Low Level	$2^{(a)(d)}$	1		(h)
	-Steam Generator Low Pressure	$3^{(a)(g)}$	1		(i)
	-Steam Generator Differential Pressure	$3^{(a)(g)}$	1		(i)

a A and B actuation circuits each have 4 channels.

b Auto removal of bypass above 1700 psia.

c Coincident high containment pressure and pressurizer pressure low signals required for initiation of containment spray.

d If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition or low level actuation position for auxiliary feedwater system within eight hours from the time of discovery of loss of operability. The remaining inoperable channel may be bypassed for 48 hours and, if an inoperable channel is not returned to operable status within this time frame, a unit shutdown must be initiated (see Specification (2)).

e Control switch on incoming breaker.

f If ~~In~~ one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from time of discovery of loss of operability. If bypassed and that channel is not returned to operable status within 48 hours from time of discovery of loss of operability, that channel must be placed in the tripped condition within the following eight hours. (See Specification (1) and exception associated with maintenance.)

g Three channels required because bypass or failure results in auxiliary feedwater actuation block in the affected channel.

h If ~~In~~ one channel becomes inoperable, that channel must be placed in the actuation condition within eight hours or bypassed condition within one hour from time of discovery of loss of operability. If bypassed and that channel is not returned to operable status within 48 hours from time of discovery of loss of operability, the channel must be placed in the low level actuation permissive condition within the following eight hours. (See Specification (1) and exception associated with maintenance.)

3.0 SURVEILLANCE REQUIREMENTS

3.0.1. Each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

3.0.2 The surveillance intervals are defined as follows:

Notation	Title	Frequency
S	Shift	At least once per 8 hours
D	Daily	At least once per 24 hours
W	Weekly	At least once per 7 days
BW	Biweekly	At least once per 14 days
M	Monthly	At least once per 31 days
Q	Quarterly	At least once per 92 days
SA	Semiannual	At least once per 184 days
A	Annually	At least once per 366 days
R	Refueling	At least once per ~~plant operating cycle~~ 18 months
P	Startup	Prior to Reactor Startup, if not completed in the previous week.

Exception to these intervals are stated in the individual Specifications.

3.0.3 The provisions of Specifications 3.0.1 and 3.0.2 are applicable to all codes and standards referenced within the Technical Specifications. The requirements of the Technical Specifications shall have precedence over the requirements of the codes and standards referenced within the Technical Specifications.

3.0.4 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specifications 3.0.1 and 3.0.2, shall constitute noncompliance with the OPERABILITY requirements for the corresponding Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

3.0 SURVEILLANCE REQUIREMENTS

3.1 Instrumentation and Control

Applicability

Applies to the reactor protective system and other critical instrumentation and controls.

Objective

To specify the minimum frequency and type of surveillance to be applied to critical plant instrumentation and controls.

Specifications

Calibration, testing and checking of instrument channels, reactor protective system and engineered safeguards system logic channels and miscellaneous instrument systems and controls shall be performed as specified in Tables 3-1 to 3-3a.

Basis

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action and a check supplements this type of built-in surveillance.

Based on the District's experience in operation of conventional power plants and on reported nuclear plant experience, a checking frequency of once-per-shift is deemed adequate for reactor and steam system instrumentation. Calibrations are performed to ensure the presentation and acquisition of accurate information.

The power range safety channels are calibrated daily against a calorimetric balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels, subject only to the "drift" errors, can be expected to remain within acceptable tolerances if recalibration is performed ~~at each refueling shutdown interval.~~ on a refueling frequency.

TABLE 3-3 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
8. Dropped CEA Indication	a. Test	R	a. Insert a negative rate of change power signal to all four Power Range Safety Channels to test alarm.
	b. Test	R	b. Insert CEA's below lower electrical limit to test dropped CEA alarm.
9. Calorimetric Instrumentation	a. Calibrate	R	a. Apply known d/p to feedwater flow sensors.
10. Control Room Ventilation	a. Test	R	a. Check damper operation for DBA mode.
	b. Test	R	b. Check control room for positive pressure.
11. Containment Humidity Detector	a. Test	R	a. Place sensor in a known high humidity atmosphere.
12. Interlocks-Isolation Valves on Shutdown Cooling Line	a. Test	R	a. Known pressure of 265 psia applied to both pressure transmitters ~~and pressure switch and operability of redundant interlock verified~~.
13. Control Room Thermometer	a. Test	R	a. Compare reading with calibrated thermometer. If not within ± 2°F, replace.

TABLE 3-4

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	Type of Measurement and Analysis	Sample and Analysis Frequency
1. Reactor Coolant		
(a) Power Operation (Operating Mode 1)	(1) Gross Radioactivity (Gamma emitters)	1 per 3 days
	(2) Isotopic Analysis for DOSE EQUIVALENT I-131	(i) 1 per 14 days (ii) 1 per 8 hours(1) ~~(1)~~ whenever the radioactivity exceeds 1.0 μCi/gm DOSE EQUIVALENT I-131. (iii) 1 sample between 2-8 hours following a thermal power change exceeding 15% of the rated thermal power ~~change exceeding 15% of the rated thermal power~~ within a 1-hour period.
	(3) $\bar{E}$ Determination	1 per 6 months(2)
	(4) Dissolved oxygen and chloride	1 per 3 days
(b) Hot Standby (Operating Mode 2)	(1) Gross Radioactivity (Gamma emitters)	1 per 3 days
Hot Shutdown (Operating Mode 3)	(2) Isotopic analysis for DOSE EQUIVALENT I-131	(i) 1 per 8 hours(1) ~~(1)~~ ~~whenever~~ the radioactivity exceeds 1.0μCi/gm DOSE EQUIVALENT I-131. (ii) 1 sample between 2-8 hours following a thermal power change exceeding 15% of the rated thermal power within a 1-hour period.
	(3) Dissolved oxygen and chloride	1 per 3 days

3.0 SURVEILLANCE REQUIREMENTS

3.3 Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance

Applicability

Applies to in-service surveillance of primary system components and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Objective

To ensure the integrity of the reactor coolant system and other components subject to inspection and testing according to ASME XI Boiler & Pressure Vessel Code.

Specifications

(1) Surveillance of the ASME Code Class 1, 2 and 3 systems, except the steam generator tubes inspection, should be covered by ASME XI Boiler & Pressure Vessel Code.

a. In-service inspection of ASME Code Class 1, Class 2, and Class 3 components and in-service testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a (g)(6)(i).

b. Surveillance of the reactor coolant pump flywheels shall be performed as indicated in Table 3-6.

c. A surveillance program to monitor radiation-induced changes in the mechanical and impact properties of the reactor vessel materials shall be maintained in accordance with 10 CFR Part 50 Appendix H.[1]

(2) Surveillance of Reactor Coolant System Pressure Isolation Valves

a. Periodic leakage testing* on each valve listed in Table 2-9 shall be accomplished prior to entering the power operation mode every time the plant is placed in the cold shutdown

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* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

The total measured leakage rate at a pressure of 60 psig shall be less than 0.75 L_a. If local leakage measurements are taken to effect repairs in order to meet 0.75 L_a acceptance criteria, these measurements shall be taken at a pressure of 60 psig.

If two consecutive Type A tests fail to meet the acceptance criteria, notwithstanding the requirements of the testing frequency, a Type A test shall be performed ~~at each refueling outage or approximately every 18 months, whichever occurs first,~~ on a refueling frequency, until two consecutive Type A tests meet the acceptance criteria, after which time the normal testing frequency schedule may be resumed.

e. Testing Frequency

A set of three Type A tests shall be performed, at approximately equal intervals during each 10 year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year in-service inspections.

The performance of Type A tests shall be limited to periods when the plant facility is non-operational and secured in the shutdown condition under administrative control and in accordance with the safety procedures defined in the license.

(4) Containment Penetrations Leak Rate Tests (Type B Tests)

a. Introduction

Type B tests are intended to detect local leaks and to measure leakage across each pressure-containing or leakage limiting boundary for the containment penetrations.

b. Test Methods

Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure of 60 psig.

Examination shall be performed by halide leak-detection method or by other equivalent test methods such as measurement of the rate of makeup required to maintain the test volume at 60 psig.

3.0 SURVEILLANCE REQUIREMENTS

3.6 Safety Injection and Containment Cooling Systems Tests

Applicability

Applies to the safety injection system, the containment spray system, the containment cooling system and air filtration system inside the containment.

Objective

To verify that the subject systems will respond promptly and perform their intended functions, if required.

Specifications

(1) Safety Injection System

System tests shall be performed ~~at each reactor~~ on a refueling ~~interval~~ frequency. A test safety feature actuation signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this portion of the test.

A second overlapping test will be considered satisfactory if control board indication and visual observations indicate all components have received the safety feature actuation signal in the proper sequence and timing (i.e., the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

(2) Containment Spray System

a. System tests shall be performed ~~at each reactor refueling interval~~ on a refueling frequency. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. At least every five years the spray nozzles shall be verified to be open.

c. The test will be considered satisfactory if:

(i) Visual observations indicate that at least 264 nozzles per spray header have operated satisfactorily.

(ii) No more than one nozzle per spray header is missing.

d. Undisturbed samples of Trisodium Phosphate Dodecahydrate (TSP) that have been exposed to the same environmental conditions as that in the mesh baskets shall be tested on a ~~routine basis once per refueling outage or at least once every 18 months by:~~ refueling frequency by:

g. Initial laboratory batch tests of charcoal adsorbers shall show $\geq$ 90% radioactive methyl iodide removal when tested under conditions of $\geq$95% relative humidity, $\geq$250°F, within $\pm$20% of design face velocity and 5 to 15 mg/m^3 inlet methyl iodide concentration. A sample shall be removed for laboratory testing ~~at each refueling outage not to exceed 18 months~~ or during the next shutdown following 4300 hours of charcoal filtering unit operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system. The results of sample tests shall show $\geq$85% radioactive methyl iodide removal under the test conditions given above.

— on a refueling frequency

Basis

The safety injection system and the containment cooling system are principal plant safeguards that are not operated during normal reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a containment spray system test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during refueling shutdowns in addition to more frequent component tests which can be performed during reactor operation.

The refueling shutdown tests demonstrate proper automatic operation of the safety injection and containment spray systems. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection actuation signals in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. (1) (2)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are to be tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent tests would result in increased wear over a long period of time. Verification

d. During refueling shutdowns the correct function of all D.C. emergency transfer switches shall be demonstrated by manual transfer of normal D.C. supply breakers at the 125 volt D.C. distribution panels.

(3) Emergency Lighting

The correct functioning of the emergency lighting system required for plant safe shutdown shall be verified at least once each year.

(4) 13.8 Kv Transmission Line

The 13.8 Kv transmission line will be energized and loaded to minimum shutdown requirements at each refueling outage following installation.

Basis

The emergency power system provides power requirements for the engineered safety features in the event of a DBA. Each of the two diesel generators is capable of supplying minimum required safety feature equipment from independent buses. This redundancy is a factor in establishing testing intervals. The monthly tests specified will demonstrate operability and load capacity of each diesel generator. These tests are conducted to meet the objectives of NRC Generic Letter 84-15 regarding the issue of reductions in cold fast starts. For this reason, the test verifying a 10 second start will be conducted from ambient conditions once per 184 days for each diesel. Other monthly tests will allow for manufacturer's recommended warm-up to reduce the mechanical stress and wear on the diesel engines. The fuel supply and various controls are continuously monitored and alarmed for off-normal conditions. ~~At approximately yearly intervals (during refueling shutdowns)~~ A Automatic starting on loss of off-site power and automatic load shedding, diesel connection, and loading will be verified on a refueling frequency. At the same intervals, capability will be verified for manual emergency control of these functions from the diesel and switch-gear rooms.

Considering system redundancy, the specified testing intervals for the station batteries should be adequate to detect and correct any malfunction before it can result in system malfunction. Batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

References

(1) USAR, Section 7.3.4.2

(2) USAR, Section 8.4.1

(3) USAR, Section 8.3.4

(4) USAR, Section 8.4.2

3.0 SURVEILLANCE REQUIREMENTS

~~3.8~~ 3.8 Main Steam Isolation Valves

Applicability

Applies to periodic testing of the main steam isolation valves.

Objective

To verify the ability of the main steam isolation valves to close upon signal.

Specifications

The operation of the main steam isolation valves shall be tested during each refueling outage to demonstrate a closure time of four seconds or less under no-flow conditions.(1)

Basis

The main steam isolation valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal will be verified at each scheduled refueling outage.

References

(1) ~~[illegible]~~ USAR, Section 10.3

3.0 SURVEILLANCE REQUIREMENTS

3.9 Auxiliary Feedwater System

Applicability

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

Objective

To verify the operability of the auxiliary feedwater (AFW) system and its ability to respond properly when required.

Specifications

(1) The position of valves necessary to ensure auxiliary feedwater flow to the steam generators shall be verified by a monthly inspection. Anytime maintenance is performed on the auxiliary feedwater system which alters valve alignments, an operator shall check that the AFW system valves are properly aligned, to ensure AFW flow to the steam generators, and a second operator shall independently verify proper valve alignment.

(2) The operability of the motor-driven auxiliary feedwater pump and the steam turbine-driven auxiliary feedwater pump shall be confirmed at least monthly.

(3) The operability of auxiliary feedwater pumps' steam generator level regulating valves HCV-1107A, HCV-1107B, HCV-1108A, HCV-1108B, and auxiliary feedwater cross-tie valve HCV-1384 shall be confirmed at least every three months.

~~(4) The capabilities of the motor-driven and turbine driven auxiliary feedwater pumps shall be verified by using local pressure indicators and flow indicators in the control room.~~ The discharge pressure will be verified to be at least 40 psig above the steam generator pressure at rated steam flow.

(4) ~~(5)~~ Following cold shutdown and prior to raising the reactor coolant temperature above 300°F, the motor-driven auxiliary feedwater pump shall be tested to verify the normal flow path for auxiliary feedwater to the steam generators.

(5) ~~(6)~~ On a refueling frequency: ~~At least once per 18 months during shutdown by~~:

a. Verify~~ing~~ that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

b. Verify~~ing~~ that each auxiliary feedwater pump starts as ~~designed~~ automatically upon receipt of each auxiliary feedwater actuation test signal.

3.0 SURVEILLANCE REQUIREMENTS

~~3.12~~ 3.13 RADIOACTIVE MATERIAL SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the NRC or an agreement State, as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals of six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

3.0 **SURVEILLANCE REQUIREMENTS**

3.14 Shock Suppressors (Snubbers)

Applicability

This specification applies to all safety-related snubbers.

Specifications

(1) All hydraulic snubbers shall be visually inspected. As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connections to the piping and anchor to verify snubber operability. In those locations where snubber movement can be manually induced without disconnecting the snubber, verify that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per functional testing acceptance criteria. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met. Visual inspections shall be performed in accordance with Table 3-14.

(2) On a refueling frequency ~~At least once per 18 months during shutdown~~ and subject to the conditions below:

(a) A representative sample (88) of hydraulic snubbers shall be functionally tested either in-place or in a bench test.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., is frozen in place, the cause will be evaluated. If the cause is a manufacturer or design deficiency, appropriate action shall be taken for snubbers of the same design subject to the same defect to determine if any more defects exist. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For any snubber(s) found locked up during normal operation or found inoperable following a seismic event, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service. If the engineering evaluation shows the components to be capable of meeting the designed service without the failed snubber, that snubber may be deleted from service per Specification 2.18(4).

(3) Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 5.10.2.m. On a refueling frequency, ~~At least once per 18 months~~ the installation and maintenance record for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

Basis

All safety snubbers shall be operable to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. The required inspection interval will be based on Table 3-14.

3.0 SURVEILLANCE REQUIREMENTS

3.16 Residual Heat Removal System Integrity Testing

Applicability

Applies to determination of the integrity of the residual heat removal systems and associated components.

Objective

To verify that the leakage from the residual heat removal system components is within acceptable limits.

Specifications

(1) a. The portion of the shutdown cooling system that is outside the containment, and the piping between the containment spray pump suction and discharge isolation valves, shall be examined for leakage at a pressure no ~~less than 250 psig.~~ This shall be performed on a refueling ~~interval.~~ frequency.

b. Piping from valves HCV-383-3 and HCV-383-4 to the suction isolation valves of the low pressure safety injection pumps and containment spray pumps and to the high pressure safety injection pumps shall be examined for leakage at a pressure no less than 82 psig. This shall be performed at the testing frequency specified in (1)a. above.

c. The portion of the high pressure safety injection (HPSI) system that is located outside the containment and downstream of the HPSI pumps shall be examined for leakage when subjected to the discharge pressure of a HPSI pump operating in the minimum recirculation mode. This test shall be performed at the frequency specified in (1)a. above. The leakage contribution from this section shall be the observed leakage from this piping at the test pressure multiplied by the square root of the ratio 1500/P, where P is the test discharge pressure (in psig) of the operating HPSI pump.

d. Visual inspection of the system's components shall be performed at the frequency specified in (1)a. above to uncover any significant external leakage to atmosphere (including leakage from valve~~s~~ stems, flanges, and pump seals). The leakage shall be measured by collection and weighing or by any other equivalent method.

(2) a. The sum of leakages from section (1)a, (1)b, and (1)c above shall not exceed 1243 cc/hour.

b. Repairs shall be made as required to maintain leakage within the acceptable limits.

4.0 **DESIGN FEATURES**

4.4 Fuel Storage

4.4.1 New Fuel Storage

The new unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less than 0.9. ~~The open grating floor below the rack and the covers above the racks, along with generous provision for drainage, precludes flooding of the new fuel storage rack.~~

New fuel may also be stored in shipping containers or in the spent fuel pool racks which have a maximum effective multiplication factor of 0.95 with Fort Calhoun Type C fuel and unborated water.

The new fuel storage racks are designed as a Class I structure.

4.4.2 Spent Fuel Storage

Irradiated fuel bundles will be stored prior to off-site shipment in the stainless steel lined spent fuel pool. The spent fuel pool is normally filled with borated water with a concentration of at least the refueling boron concentration.

The spent fuel racks are designed as a Class I structure.

Normally the spent fuel pool cooling system will maintain the bulk water temperature of the pool below 120°F. Under other conditions of fuel discharge, the fuel pool water temperature is maintained below 140°F.

The spent fuel racks are designed and will be maintained such that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) assuming the pool is flooded with unborated water. The racks are divided into 2 regions. Region 1 racks are surrounded by Boraflex; Region 2 racks have no poison. Acceptance criteria for fuel storage in Regions 1 and 2 are delineated in Section 2.8 of these Technical Specifications.

The new fuel storage rack is located 18'-9" above the main floor of Room 25A which provides for adequate drainage and precludes flooding of the new fuel storage rack.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Manager - Fort Calhoun Station shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

5.2 Organization

5.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the USAR.

b. The Manager - Fort Calhoun Station shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Senior Vice President - shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Plant Staff

The plant staff organization shall be as described in Chapter 12 of the USAR and shall function as follows:

a. The minimum number and type of licensed and unlicensed operating personnel required onsite for each shift shall be as shown in Table 5.2-1.

Responsibilities

5.5.1.6 The Plant Review Committee shall be responsible for:

a. Review of (1) Administrative Controls Standing Orders and changes thereto, (2) procedures required by Specification 5.8 and requiring a 10 CFR 50.59 safety evaluation, and (3) proposed changes to procedures required by Specification 5.8 and requiring a 10 CFR 50.59 safety evaluation;

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes to the Core Operating Limits Report.

e. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

f. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Division Manager - Nuclear Operations and to the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews and investigations and reports thereon as requested by the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee.

i. Review of the Site Security Plan and implementing procedures and shall submit recommended changes to the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee.

j. Review of the Site Emergency Plan and implementing procedures and shall submit recommended changes to the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee.

k. Review of all Reportable Events.

Authority

5.5.1.7 The Plant Review Committee shall:

a. Recommend in writing to the Manager - Fort Calhoun Station approval or disapproval of items considered under 5.5.1.6(a) through (e) above.

5.5.1.7 b. Render determinations in writing with regard to whether or not each item considered under 5.5.1.6(b) through (f) above constitutes an unreviewed safety question.

c. Provide immediate written notification to the Division Manager - Nuclear Operations and the Safety Audit and Review Committee of disagreement between the Plant Review Committee and the Manager - Fort Calhoun Station; however, the Manager - Fort Calhoun Station shall have responsibility for resolution of such disagreements pursuant to 5.1.1 above.

Records

5.5.1.8 The Plant Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Division Manager - Nuclear Operations and ~~Chairman~~ Chairperson of the Safety Audit and Review Committee.

5.5.2 Safety Audit and Review Committee (SARC)

Function

5.5.2.1 The Safety Audit and Review Committee shall function to provide the independent review and audit of designated activities in the areas of:

a. nuclear power plant operation
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance

Composition

5.5.2.2 The Safety Audit and Review Committee shall be composed of:

Chairperson
~~Chairman:~~ Division Manager - Nuclear Services
Member: Senior Vice President
Member: Division Manager - Nuclear Operations
Member: Division Manager - Production Engineering
Member: Manager - Fort Calhoun Station
~~Member: Manager - Radiological Services~~
Member: ~~Qualified Consultants as Required and as Determined by SARC Chairman~~ Other qualified OPPD personnel or consultants as required and as determined by the SARC Chairperson

Member: Vice President

5.0 ADMINISTRATIVE CONTROLS

Alternates

5.5.2.3 Alternate members shall be appointed in writing by the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee to serve on a temporary basis; however, no more than two alternates may participate in the Safety Audit and Review Committee activities at any one time.

Consultants

5.5.2.4 Consultants shall be utilized as determined by the Safety Audit and Review Committee ~~Chairman~~ Chairperson to provide expert ~~advise~~ advice to the Safety Audit and Review Committee.

Meeting Frequency

5.5.2.5 The Safety Audit and Review Committee shall meet at least once every six months.

Quorum

5.5.2.6 A quorum of the Safety Audit and Review Committee shall consist of the ~~Chairman~~ Chairperson or his designated alternate and a majority of the Safety Audit and Review Committee members including alternates. No more than a minority of the quorum shall have line responsibility for the operation of the nuclear plant.

Review

5.5.2.7 The Safety Audit and Review Committee shall review:

a. The safety evaluations for 1) procedures, equipment or systems and 2) tests or experiments completed under the provision of section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in section 50.59, 10 CFR.

5.5.2.7 c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All Reportable Events.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Plant Review Committee.

The ~~Chairman~~ Chairperson of the Safety Audit and Review Committee (SARC) may designate subgroups, special working committees, or audit teams as he deems necessary in order to carry out the responsibilities of the SARC. These subgroups, committees, or audit teams will perform the SARC responsibilities and report on their activities for review at the next regularly scheduled SARC meeting following any group's action.

Audit

5.5.2.8 Audits of facility activities shall be performed under the cognizance of the Safety Audit and Review Committee. These audits shall encompass:

a. The conformance of facility operation to ~~all~~ provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of ~~all~~ actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of ~~all~~ activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per two years.

5.5.2.8 e. The Fort Calhoun Station Emergency Plan and implementing procedures at least once every twelve months.

f. The Site Security Plan and implementing procedures at least once every twelve months.

g. The Safeguards Contingency Plan and implementing procedures at least once every twelve months.

h. The Radiological Effluent Program including the Radiological Environmental Monitoring Program and the results thereof, the Offsite Dose Calculation Manual and implementing procedures, and the Process Control Program for the solidifications of radioactive waste at least once per 2 years.

i. Any other area of facility operation considered appropriate by the Safety Audit and Review Committee or the Senior Vice President.

[INSERT SPECIFICATION 5.5.3]

~~Authority~~

5.5.2.9 The Safety Audit and Review Committee shall report to and advise the Senior Vice President on those areas of responsibility specified in Sections 5.5.2.7 and 5.5.2.8.

Records

5.5.2.10 Records of Safety Audit and Review Committee activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each Safety Audit and Review Committee meeting shall be prepared, approved and forwarded to the Senior Vice President within ~~14~~ 30 days following each meeting.

b. Reports of reviews encompassed by Section 5.5.2.7e, f, g, h, and i above shall be prepared, approved and forwarded to the Senior Vice President within ~~14~~ 30 days following completion of the review.

c. Audit reports encompassed by Section 5.5.2.8 above shall be forwarded to the Senior Vice President and to the responsible management positions designated by the Safety Audit and Review Committee within 30 days after completion of the audit.

MOVE TO SPECIFICATION 5.5.2.8

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.5.3 Fire Protection Inspection~~

j. ~~a.~~ An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm. ~~The audit and inspection program responsibility shall rest with the Safety Audit and Review Committee~~.

k. ~~b.~~ An inspection and audit of the fire protection and loss prevention program by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

5.6 Reportable Event Action

5.6.1 The following actions shall be taken in the event of a REPORTABLE EVENT:

a. The Commission shall be notified pursuant to the requirements of 10 CFR 50.72, if applicable.

b. Each Reportable Event shall be reviewed by the Plant Review Committee and submitted to the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee and the Division Manager - Nuclear Operations.

c. Submit reports of Reportable Events pursuant to the requirements of Specification 5.9.2.

5.7 Safety Limit Violation

5.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within 1 hour.

b. The Safety Limit Violations shall be reported to the Division Manager - Nuclear Operations and the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee (SARC) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee and the Division Manager - Nuclear Operations within 14 days of the violation.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the minimum requirements of sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33 except as provided in 5.8.2 and 5.8.3 below.

5.8.2 Each procedure of Specification 5.8.1, and changes thereto, and any other procedure or procedure change that the Manager - Fort Calhoun Station determines to affect nuclear safety, shall be reviewed and approved as described below, prior to implementation.

5.9.1 Continued

work and job functions,[3/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. <u>Monthly Operating Report</u>. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U. S. Nuclear Regulatory Commission, Document Control Desk, Mail Station P1-137, Washington, D. C. 20555, with a copy to the appropriate Regional Office~~, to arrive~~ no later than the fifteenth of each month following the calendar month covered by the report. This monthly report shall also include a statement regarding any challenges or failures to the pressurizer power operated relief valves or safety valves occurring during the subject month.

5.9.2 <u>Reportable Event</u>

A Licensee Event Report (LER) shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Mail Station P1-137, Washington, D. C. 20555 with a copy to Region IV of the NRC, within 30 days after discovery of any event meeting the requirements of 10 CFR 50.73.

3/ This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

ATTACHMENT B

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATIONS

DISCUSSION AND JUSTIFICATION

The Omaha Public Power District (OPPD) proposes to revise the Fort Calhoun Station Unit No. 1 Technical Specifications to implement administrative changes. The following is a list of pages affected and the paragraphs where the proposed changes are described.

Page iii	- See paragraph 25	Page 3-56	- See paragraph 1
Page 4	- See paragraph 3	Page 3-60	- See paragraphs 1 & 16
Page 1-4	- See paragraph 2	Page 3-61	- See paragraph 17
Page 1-8	- See paragraph 2	Page 3-62	- See paragraphs 1 & 18
Page 2-15a	- See paragraph 2	Page 3-76	- See paragraph 19
Page 2-18	- See paragraph 4	Page 3-77	- See paragraph 1
Page 2-20	- See paragraph 5	Page 3-79	- See paragraph 1
Page 2-22	- See paragraph 6	Page 3-84	- See paragraphs 1 & 20
Page 2-57e	- See paragraph 7	Page 4-4	- See paragraph 21
Page 2-58	- See paragraph 8	Page 5-1	- See paragraph 22
Page 2-62	- See paragraphs 9 & 10	Page 5-4	- See paragraph 23
Page 2-64	- See paragraph 9	Page 5-5	- See paragraph 23
Page 2-66	- See paragraph 11	Page 5-6	- See paragraphs 23 & 24
Page 2-68a	- See paragraph 12	Page 5-7	- See paragraphs 23 & 25
Page 3-0a	- See paragraph 1	Page 5-8	- See paragraphs 23 & 25
Page 3-1	- See paragraph 1	Page 5-8a	- See paragraph 25
Page 3-15	- See paragraph 13	Page 5-9	- See paragraphs 23 & 25
Page 3-18	- See paragraph 14	Page 5-12	- See paragraph 26
Page 3-21	- See paragraph 15		
Page 3-40	- See paragraph 1		
Page 3-54	- See paragraph 1		

1. In order to provide consistency throughout the Technical Specifications with a defined term, it is proposed to revise surveillance requirements which state an "18 month," or "18 month during shutdown" interval to state that these surveillances are conducted on a "refueling frequency." Refueling frequency is defined in Specification 3.0.2 as at least once per plant operating cycle. To ensure that the 18 month interval is adequately stated in the Technical Specifications, it is proposed that Specification 3.0.2 be revised to reflect the CE Standard Technical Specification definition for refueling frequency which is at least once per 18 months. The following Specifications are affected by this proposed change.

Page 3-0a	Specification 3.0.2
Page 3-1	Basis of Specification 3.1
Page 3-40	Specification 3.5(3)d.
Page 3-54	Specifications 3.6(1), 3.6(2)a. and 3.6(2)d.
Page 3-56	Specification 3.6(2)g.
Page 3-60	Basis of Specification 3.7
Page 3-62	Specification 3.9(6)
Page 3-77	Specification 3.14(2)
Page 3-79	Specification 3.14(3)
Page 3-84	Specification 3.16(1)a.

2. Page 1-4, 1-8, and 2-15a

The bases to Specifications 1.2 and 1.3 are being revised to be consistent with Table 1-1, Item No. 5, and Specification 2.1.6 is being revised to add discussion concerning the presence of water-filled loop seals upstream of the pressurizer safety valves.

Technical Specification 2.1.6(1) requires that the pressurizer safety valves be operable with their lift settings adjusted to ensure valve opening at 2500 psia $\pm$ 1% and 2545 psia $\pm$ 1%. The setpoints are established in accordance with the ASME code as required by Specification 3.3(1). The ASME code requires that safety valves whose design basis is to relieve steam pressure be set using steam inlet conditions. Upstream of the pressurizer safety valves are water-filled loop seals designed to reduce leakage of non-condensible gases through the valves. Since the ASME code requires that the setpoints be established using steam, the presence of the water-filled loop seal could potentially influence the pressure at which these valves open.

To address this issue, OPPD completed Engineering Analysis EA-FC-92-066 which verified that the reactor coolant system could withstand an overpressure transient with a safety valve setpoint deviation of +6% and be within the results contained in the Updated Safety Analysis Report. In addition, it was determined that a safety valve setpoint deviation as low as -4% would not cause unnecessary challenges to safety systems. The basis of Specification 2.1.6 is being revised to incorporate this additional information concerning the presence of the water filled loop seals and potential setpoint deviations. In addition, the bases of Specification 1.2 and 1.3 are being revised to indicate that the reactor high pressure trip is set at less than or equal to 2400 psia which is consistent with the requirements of Specification 1.2, Table 1-1, Item No. 5, and that the PORVs setpoint is consistent with the reactor high pressure trip.

The wording "steam system safety valves," is also being revised in the basis of Specification 1.2 to "main steam safety valves" to be consistent with wording implemented in Amendment 146.

The word "or" is being corrected to the word "of" in the basis of Specification 2.1.6.

3. Page 4

The definition of Channel Check contained on page 4 is being revised to correct a typographical error. The word "behaviour" is misspelled and is being corrected to read "behavior."

4. Page 2-13

Specification 2.2(2)d1. is being clarified by adding valve LCV-218-3 to the equipment required to be operable when the required volume of boric acid may be combined between Boric Acid Storage Tanks (BAST) CH-11A and CH-11B. Specifications 2.2(2)d2. through 2.2(2)d4. state the requirements when LCV-218-3, CH-11B, and CH-11A are inoperable.

5. Page 2-20

Specification 2.3(1)c. is being revised to clarify that it is required that all four safety injection tanks have a tank "level" of at least 116 inches. The current specification requirement to have a tank "liquid" of at least 116 inches is not correct grammar and is being corrected.

Specification 2.3(1)e. and 2.3(1)f. are being clarified to indicate that it is required to maintain at least one low pressure safety injection pump and one high pressure safety injection pump on each "associated 4160 V engineered safety feature" bus. These specifications ensure that there is electrical independence for the pumps. There are two low pressure safety injection pumps, one powered from each 4160 V bus. There are three high pressure safety injection pumps powered from three 480 V buses. Two of the 480 V buses are independently powered, one from each 4160 V bus, and one is a "swing bus" which could be powered from either of the 4160 V buses.

Specification 2.3(1)f. ensures the minimum requirements of maintaining two high pressure safety injection pumps with electrical independence. Specification 2.3(1)j. ensures maintaining two high pressure safety injection pumps with independent suction sources. This change only clarifies that the buses stated in Specifications 2.3(1)e. and 2.3(1)f. are the 4160 V buses and not the 480 V buses, thereby requiring that a minimum of two independent high and low pressure safety injection pumps are operable.

6. Page 2-22

It is proposed to revise the Basis of Specification 2.3, "Emergency Core Cooling System," to delete the reference to low temperature/low power physics testing. This low temperature testing refers to the one-time testing conducted at 200°F which was performed during the initial startup (Cycle 1). Specification 2.3 requires minimum ECCS equipment be operable before the reactor can be made critical. There are no special test exceptions stated in Specification 2.3. Therefore, these requirements also apply when the reactor is made critical during low power physics testing and the statement in the basis which indicates that ECCS is not required is incorrect and is being deleted.

7. Page 2-57e

The basis to Specification 2.10.4 is being revised to delete the specific steps of how to measure RCS flow by using reactor coolant pump differential pressure. The specific steps on conducting this test are appropriately included in procedures, and it is inappropriate to include the specific steps in the basis of a specification. The phrase "that will" is being revised to "may" to indicate that pump differential pressure is not the only available method for determining RCS flow.

8. Page 2-58

Reference (1) on page 2-58 is being revised from FSAR, Section 14.18 to the current nomenclature for this document which is USAR (Updated Safety Analysis Report) Section 14.18.

9. Pages 2-62 and 2-64

It is proposed to revise Specification 2.14(3) "Engineered Safety Features System Initiation Instrumentation Settings - Containment High Radiation (Air Monitoring)" and Table 2-1, "Engineered Safety Features System Initiation Instrument Setting Limits" to correct inconsistences between Technical Specification 2.14 and the Offsite Dose Calculation Manual (ODCM). Radiation Monitors RM-050 and RM-051 are process monitors; however, they may be considered to be effluent monitors when monitoring the Auxiliary Building Exhaust Stack. The ODCM is utilized to control effluent radiation monitor setpoints but not process radiation monitor setpoints. Therefore, \ is proposed that isolation function setpoints for effluent monitors be calculated in accordance with the ODCM and isolation function setpoints for process radiation monitors be calculated in accordance with the applicable Chemistry Manual calibration procedure.

10. Page 2-62

It is proposed to revise the Basis of Technical Specification 2.14(5) to delete the specific value for the valve stroke time. The basis for this specification is that the valves close in sufficient time to ensure adequate net positive suction head is available to the safety injection pumps. Deleting the specific stroke time will make the basis of Specification 2.14(5) consistent with the remainder of the Technical Specifications which do not state specific valve stroke times.

The reference to the FSAR loss of coolant accident analysis described in the basis of 2.14(5) is being revised to the current nomenclature for this document which is the USAR.

11. Page 2-66

Specification 2.15(2) is being revised to correct a typographical error. The statement "...channel has not been stored to operable status," is being corrected to read "...channel has not been restored to operable status." The word "store" is being replaced by the correct word "restored."

Specification 2.15(3) is being revised to correct a typographical error. The statement '...falls below the limits given in the columns entitled "Minimum Operable Channels" of "Minimum Degree of Redundancy,' is being corrected to read '...falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy.' The word "of" is being replaced by the correct word "or."

12. Page 2-68a

Table 2-3, Footnotes f and h are being revised to correct typographical errors. The statement "In one channel becomes inoperable..." is being corrected to read "If one channel becomes inoperable..." The word "In" is being replaced by the correct word "If."

13. Page 3-15

Table 3-3, Item 12, Surveillance Method, is being clarified to indicate that the known pressure is applied to two separate pressure transmitters. The known pressure cannot be applied to the pressure switch. The "redundant interlock" discussed in this item is a separate pressure transmitter in a separate instrument loop. The proposed wording only clarifies where the known pressure is applied.

14. Page 3-18

Table 3-4, Item 1.(a)(2)(ii), is being revised to make the "(1)" a superscript as this item applies to Note (1) contained on page 3-19.

Table 3-4, Item 1.(a)(2)(iii), is being revised to delete redundant wording. The phrase "change exceeding 15% of the rated thermal power" is included twice in the specification.

Table 3-4, Item 1.(b)(2)(i), is being revised to state the sample frequency of 1 per "8" hours and to make the "(1)" a superscript as this requirement applies to Note (1) contained on page 3-19.

These items were inadvertently incorporated in Amendment 133.

15. Page 3-21

It is proposed to revise Specification 3.3(2)a, to add an asterisk that was inadvertently deleted in an earlier amendment. The deletion occurred in Amendment 104 when Specification 3.3 was reorganized to incorporate a separate specification for steam generator tube inspections.

16. Page 3-60

Specification 3.7(3) is being revised to clarify that the emergency lighting system required to be surveillance tested in accordance with this specification is the emergency lighting required for plant safe shutdown.

17. Page 3-61

Specification 3.8 is being revised to correct a typographical error and to update the reference. The numbering of Specification 3.8 is incorrectly identified as "3,8", the comma is being replaced by a period. Reference (1) is being revised from "FSAR" to the current nomenclature for this reference which is "USAR."

18. Page 3-62

Specification 3.9(2) and Specification 3.9(4) are being combined. As presently written, Specification 3.9(4) is not a surveillance, but is the acceptance criteria for the surveillance required by 3.9(2), therefore, it is appropriate for these two specifications to be combined. It is also proposed that the statement concerning the location of where readings will

proposed that the statement concerning the location of where readings will be taken be deleted as it is unnecessary. Specifications 3.9(5) and 3.9(6) are renumbered to reflect the deletion of 3.9(4). Specification 3.9(6)a. and 3.9(6)b. are being revised to change the word "verifying" to "verify." The phrase "at least" is being added to Specification 3.9(4) to clarify that the acceptance criteria for pump pressure is at least 40 psig above the steam generator pressure at rated steam flow.

19. Page 3-76

It is proposed to revise the numbering of Specification 3.12, "Radioactive Material Sources Surveillance," to correct a typographical error. The correct number is "Specification 3.13," as "Specification 3.12" is, "Radioactive Waste Disposal System."

20. Page 3-84

It is proposed to revise the word "valves" contained in Specification 3.16(1)d. to the correct word "valve."

21. Page 4-4

Specification 4.4.1 is being clarified. The current discussion implies that the floor below the new fuel rack is made entirely of open grating, which is not true. The proposed change would clarify that the design basis of the storage area for new fuel is to preclude flooding.

22. Page 5-1

Specification 5.2.2.a. is being revised to add a line which was inadvertently deleted in Amendment 132. The requirement states that "The minimum number and type of licensed and unlicensed operating Table 5.2-1." The requirement should state, "The minimum number and type of licensed and unlicensed operating personnel required onsite for each shift shall be as shown in Table 5.2-1." The line "personnel required onsite for each shift shall be as shown in," is being added to correct this specification.

23. Pages 5-4, 5-5, 5-6, 5-7, 5-8, and 5-9

It is proposed to revise Specification 5.5 to reflect organizational changes, title changes and to revise the submittal of Safety Audit and Review Committee reports to the Senior Vice President from 14 days to 30 days. The title changes involve: revising the title "Chairman" to "Chairperson," clarifying the SARC membership by allowing additional technical experts to be members at the discretion of the SARC Chairperson, deleting the Manager - Radiological Services, and adding the Vice President as a member. The timeframe for submittal of SARC reports to the Senior Vice President is being revised because the Senior Vice President is a member of the SARC as specified in Specification 5.5.2.2. Therefore, this position is knowledgeable of actions taken by this committee, and 30 days is more than adequate for submittal of the written report. Thirty days is consistent with NUREG-1432 Specification 5.5.2.c.

24. Page 5-6

Specification 5.5.2.4 is being revised to correct a typographical error. The word "advise" is being replaced with the word "advice" to correct the error.

25. Pages 5-7, 5-8, 5-9, and Page iii of the Table of Contents

It is proposed to revise Specification 5.5.2.8 and Specification 5.5.3 to be more consistent with the CE Standard Technical Specification 6.5.2.8 (NUREG-0212 R2). Specification 5.5.3 will be deleted and incorporated into 5.5.2.8. Since Specification 5.5.3.a is being moved to 5.5.2.8 the sentence stating that this audit is under the cognizance of the SARC is no longer required and is being deleted. The audit schedule for review of the Emergency Plan and Safeguards Contingency Plan are being maintained at a 12 month interval consistent with 10 CFR 50.54. The requirement to audit the Radiological Effluent Program, Specification 5.5.2.8.h, is being maintained as it is included in NUREG-1432. Page iii of the Table of Contents is being revised to reflect the deletion of Specification 5.5.3.

26. Page 5-12

It is proposed to revise Specification 5.9.1.c to indicate that the Monthly Operating Report will be submitted "no later than the fifteenth of each month" instead of "to arrive no later than the fifteenth" of each month. This requirement is consistent with CE Standard Technical Specification 6.9.1.6 (NUREG-0212 R2).

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION:

The proposed changes do not involve significant hazards considerations because operation of Fort Calhoun Station Unit No. 1 in accordance with these changes does not:

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

The proposed changes include: administrative changes to correct typographical errors and references, make the specifications consistent, provide clarifications, make changes consistent with organizational changes, or with the CE Standard Technical Specifications.

The clarification to the basis of Specification 2.1.6 provides a discussion on: the presence of water filled loop seals, the potential effects the loop seal may have on the setpoint deviation of the safety valves, and that any effect is within the results of the Updated Safety Analysis Report.

The clarification to Specification 2.2(2)d1. provides an additional requirement to maintain valve LCV-218-3 operable which is consistent with the intent of the specification in that the valve must be operable to maintain the required flow path from the Safety Injection and Refueling Water (SIRW) tank.

The clarification to Specification 2.3(1) states which electrical buses the safety injection pumps are powered through and is consistent with the Updated Safety Analysis Report, Section 14.15, which assumes that only one full capacity high pressure pump and one full capacity low pressure pump are available during a Loss of Coolant Accident.

The clarification to the basis of Specification 2.14 only deletes the reference to the specific time for a valve to open.

The clarification to Specification 3.7(3) adds verbiage to state that the emergency lighting system required to be tested by this specification is the emergency lighting system required to achieve a plant safe shutdown.

The proposed changes are administrative in nature and are consistent with the assumptions or results stated in the Updated Safety Analysis Report; therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed administrative changes correct typographical errors and references, and implement changes to make the Specifications consistent. No new or different operation of plant equipment is proposed. No new or different action statements are proposed. Therefore, the proposed changes do not create the possibility of a new or different type of accident.

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

The proposed administrative changes correct typographical errors and references, and implement changes to make the Specifications consistent. The clarifications being proposed are within the assumptions or results as stated in the Updated Safety Analysis Report; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Therefore, based on the above considerations, it is OPPD's position that this proposed amendment does not involve significant hazards considerations as defined by 10 CFR 50.92 and the proposed changes will not result in a condition which significantly alters the impact of the Station on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(e)(9) and pursuant to 10 CFR 51.22(b) no environmental assessment need be prepared.