TECHNICAL SPECIFICATIONS

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Amendment No. ~~32,38,52,54,57,67,80,81, 86,146,~~ 152

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2.0 LIMITING CONDITIONS FOR OPERATION

2.5 Steam and Feedwater Systems

Applicability

Applies to the operating status of the steam and feedwater systems.

Objective

To define certain conditions for the steam and feedwater system necessary to assure adequate decay heat removal.

Specifications

The reactor coolant shall not be heated above 300°F unless the following conditions are met:

(1) The motor driven auxiliary feedwater pump is operable. The reactor shall not be made critical unless the steam driven auxiliary feedwater pump is operable. During modes 1 and 2, one auxiliary feedwater pump may be inoperable for up to 24 hours, provided that the redundant component shall be tested to demonstrate operability.

(2) A minimum of 55,000 gallons of water in the emergency feedwater storage tank and a backup water supply to the emergency feedwater storage tank ~~from the Missouri River by the fire water system~~ shall be available.

(3) All valves, interlocks and piping associated with the above components required to function during accident conditions are operable. Manual valves that could interrupt auxiliary feedwater flow to the steam generators shall be locked in the required position to ensure a flow path to the steam generators.

(4) The main steam stop valves are operable and capable of closing in four seconds or less under no-flow conditions.

Basis

A reactor shutdown from power requires a removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as long as feedwater to the steam generator is available. Normally, the capability to supply feedwater to the steam generators is provided by operation of the turbine cycle feedwater system. In the unlikely event of complete loss of electrical power to the station, decay heat removal is by steam discharge to the atmosphere via the main steam safety and atmospheric dump valves. Either auxiliary feedwater pump

(6) Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

(7) When irradiated fuel is being handled in the auxiliary building, the exhaust ventilation from the spent fuel pool area will be diverted through the charcoal filter.

DELETED

(8) ~~Prior to initial core loading and prior to refueling operations, a complete check out, including a load test, shall be conducted on fuel handling cranes that will be required during the refueling operation to handle spent fuel assemblies.~~

(9) A minimum of 23 feet of water above the top of the core shall be maintained whenever irradiated fuel is being handled.

(10) Storage in Region 1 and Region 2 of the spent fuel racks shall be restricted to fuel assemblies having initial enrichment less or equal to 4.2 weight percent of U-235.

(11) Storage in Region 2 of the spent fuel racks shall be restricted to those assemblies whose parameters fall within the "acceptable" area of Figure 2-10. Storage in the peripheral cells of Region 2 shall be restricted to those assemblies whose parameters fall within the noted area of Figure 2-10.

(12) A minimum boron concentration of 100 ppm shall be maintained in the Spent Fuel Pool whenever storing unirradiated fuel in the Spent Fuel Pool.

If any of the above conditions are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions, and no operations that may change the reactivity of the core shall be made.

A spent fuel assembly may be transferred directly from the reactor core to the spent fuel pool Region 2 provided the independent verification of assembly burnups has been completed and the assembly burnup meets the acceptance criteria identified in Technical Specification Figure 2-10.

Movement of irradiated fuel from the reactor core shall not be initiated before the reactor core has been subcritical for a minimum of 72 hours if the reactor has been operated at power levels in excess of 2% rated power.

Bases

and

The equipment and general procedures to be utilized during refueling operations are discussed in the USAR. Detailed instructions, the above specifications, ~~and the design of the fuel handling equipment incorporating built-in interlocks and safety features~~ provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.~~[1]~~ Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The shutdown cooling pump is used to maintain a uniform boron concentration.

The shutdown margin as indicated will keep the core subcritical even if all CEA's were withdrawn from the core. During refueling operations, the reactor refueling cavity is filled with approximately 250,000 gallons of borated water. The boron concentration of this water (of at least the refueling boron concentration) is sufficient to maintain the reactor subcritical by more than 5%, including allowance for uncertainties, in the cold condition with all rods withdrawn.[2](1) Periodic checks of refueling water boron concentration ensures the proper shutdown margin. Communication requirements allow the control room operator to inform the refueling machine operator of any impending unsafe condition detected from the main control room board indicators during fuel movement.

~~In addition to the above engineered safety features, interlocks are utilized during refueling operations to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. In addition, interlocks on the auxiliary building crane will prevent the trolley from being moved over storage racks containing irradiated fuel, except as necessary for the handling of fuel.~~ The restriction of not moving fuel in the reactor for a period of 72 hours after the power has been removed from the core takes advantage of the decay of the short half-life fission products and allows for any failed fuel to purge itself of fission gases, thus reducing the consequences of fuel handling accident.

The ventilation air for both the containment and the spent fuel pool area flows through absolute particulate filters and radiation monitors before discharge at the ventilation discharge duct. In the event the stack discharge should indicate a release in excess of the limits in the technical specifications, the containment ventilation flow paths will be closed automatically and the auxiliary building ventilation flow paths will be closed manually. In addition, the exhaust ventilation ductwork from the spent fuel storage area is equipped with a charcoal filter which will be manually put into operation whenever irradiated fuel is being handled.~~[1]~~

The basis for the 100 ppm boron concentration requirement with Boral poisoned storage racks is to maintain the k_{eff} below 0.95 in the event a misplaced unirradiated fuel assembly is located next to a spent fuel assembly. A misplaced unirradiated fuel assembly at 4.2 w/o enrichment condition, in the absence of soluble poison, may result in exceeding the design effective multiplication factor. Soluble boron in the Spent Fuel Pool water, for which credit is permitted under these conditions, would assure that the effective multiplication factor is maintained substantially less than the design condition. The boron concentration is periodically sampled in accordance with Specification 3.2.

References

~~(1) USAR, Section 9.5~~

(1) (2) USAR, Section 9.5.1.2

2.0 <u>LIMITING CONDITIONS FOR OPERATION</u>
2.10 <u>Reactor Core</u> (Continued)
2.10.4 <u>Power Distribution Limits</u> (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 be used to measure flow.

(next page is 2-59)

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) USAR, Section 14.18

3.0 SURVEILLANCE REQUIREMENTS

3.2 Equipment and Sampling Tests (continued)

The Safety Injection (SI) pump room air treatment system consists of charcoal adsorbers which are installed in normally bypassed ducts. This system is designed to reduce the potential release of radioiodine in SI pump rooms during the recirculation period following a DBA. The in-place and laboratory testing of charcoal adsorbers will assure system integrity and performance.

Pressure drops across the combined HEPA filters and charcoal adsorbers, of less than 9 inches of water for the control room filters (VA-64A & VA-64B) and of less than 6 inches of water for each of the other air treatment systems will indicate that the filters and adsorbers are not clogged by amounts of foreign matter that would interfere with performance to established levels. Operation of each system for 10 hours every month will demonstrate operability and remove excessive moisture build-up in the adsorbers.

The hydrogen purge system provides the control of combustible gases (hydrogen) in containment for a post-LOCA environment. The surveillance tests provide assurance that the system is operable and capable of performing its design function. VA-80A or VA-80B is capable of controlling the expected hydrogen generation (67 SCFM) associated with 1) Zirconium - water reactions, 2) radiolytic decomposition of sump water and 3) corrosion of metals within containment. The system should have a minimum of one blower with associated valves and piping (VA-80A or VA-80B) available at all times to meet the guidelines of Regulatory Guide 1.7 (1971).

~~The requirements in 2.9.1(2)f(iii) to automatically isolate the hydrogen purge system by the stack radiation monitors does not apply in that the isolation valves do not have automatic isolation capability.~~

If significant painting, fire or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals or foreign materials, testing will be performed to confirm system performance.

Demonstration of the automatic and/or manual initiation capability will assure the system's availability.

References

USAR, Section 9.10

TABLE 3-5

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency	USAR ~~FSAR~~ Section Reference
1.	Control Element Assemblies	Drop times of all full-length CEA's	Prior to reactor criticality after each ~~Each refueling operation~~ removal of the reactor vessel closure head	7.5.3
2.	Control Element Assemblies	Partial movement of all CEA's (Minimum of 6 in)	~~Every two weeks~~ BW	7
3.	Pressurizer Safety Valves	Set Point	~~Once each refueling outage~~ R	7
4.	Main Steam Safety Valves	Set Point	~~Each refueling outage~~ R	4
5.	DELETED ~~Refueling System Interlocks~~	~~Functioning~~	~~Prior to refueling outage~~	~~9.5.6~~
6.	DELETED			
7.	DELETED			
8.	Reactor Coolant System Leakage	Evaluate	~~Daily*~~ D*	4
9.	Diesel Fuel Supply	Fuel Inventory	~~Daily~~ D	8.4
10a.	Charcoal and HEPA Filters for Control Room	1. In-Place Testing** Charcoal adsorbers and HEPA filter banks shall be leak tested and show $\geq$99.95% Freon (R-11 or R-112) and cold DOP particulates removal, respectively.	On a refueling frequency ~~Each refueling shutdown not to exceed 18 months~~ or ~~after~~ every 720 hours of system operation or after each complete or partial replacement of the charcoal adsorber/HEPA filter banks, or after any major structural maintenance on the system housing ~~and~~ or following significant painting, fire or chemical releases in a ventilation zone communicating with the system.	9.10

* Whenever the system is at or above operating temperature and pressure.

** Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

TABLE _ 5
(Continued)

	Test	Frequency	USAR ~~FSAR~~ Section Reference
10a. (continued)	2. Laboratory Testing** a. Initial batch tests of activated charcoal shall show 99.825% radioactive methyl iodide removal when tested under conditions of ≥ 70% relative humidity, ≥ 176°F (80°C) 1.5 to 2.0 mg/m^3 inlet methyl iodide concentration at a face velocity of 40± 1.6 FPM (12.2 ± 0.5 m/min) and at a bed depth of 4 inches (101.6 mm).	Prior to initial loading in filter unit.	
	b. Activated charcoal cells shall be replaced or tested. The test results shall show ≥ 99.825% methyl iodide removal when tested under conditions of ≥ 70% relative humidity, ≥ 176°F (80°C), at a face velocity of 40 ± 1.6 FPM (12.2 ± 0.5 m/min) and at a bed depth of 4 inches (101.6 mm).	On a refueling frequency ~~Each refueling shutdown not to exceed 18 months~~ or ~~after~~ every 720 hours of system operation or ~~and~~ following significant painting, fire or chemical release in a ventilation zone communicating with the system.	
	3. Overall System Operation a. Each circuit shall be operated.	Ten hours every month	
	b. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 9 inches of water at system design flow rate.	~~At least once per plant operating cycle.~~ R	
	c. Fan shall be shown to operate within ± 10% design flow.	~~At least once per plant operating cycle.~~ R	
	4. Automatic and manual initiation of the system shall be demonstrated.	~~At least once per plant operating cycle.~~ R	

** Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

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TABLE 3-5 (Continued)

			Test	Frequency	USAR Section Reference
10b.	Charcoal Adsorbers for Spent Fuel Storage Pool Area	1.	In-Place Testing** Charcoal adsorbers shall be leak tested and shall show $\geq$99% Freon (R-11 or R-112) removal.	On a refueling frequency ~~Each month refueling shutdown not to exceed 18 months~~ or ~~after~~ every 720 hours of system operation, or after each complete or partial replacement of the charcoal adsorber bank, or after any major structural maintenance on the system housing or ~~and~~ following significant painting, fire or chemical release in a ventilation zone communicating with the system.	6.2 9.10
		2.	Laboratory Testing		
			a. Initial batch tests of all charcoal adsorbers shall show $\geq$99% elemental iodine removal when tested under conditions of $\geq$95% R.H., $\geq$125°F, 5 to 10 mg/m^3 inlet elemental iodine concentration and at the face velocity within $\pm$20% of system design.	Prior to initial loading in the filter unit.	
			b. The carbon sample test results shall show $\geq$90% elemental iodine removal, under conditions of $\geq$95% R.H., $\geq$125°F, 5 to 10mg/m^3 inlet elemental concentration and within 20% of design face velocity.	On a refueling frequency ~~Each refueling shutdown not to exceed 18 months~~ or ~~after~~ every 720 hours of system operation, or ~~and~~ following significant painting, fire or chemical release in any ventilation zone communicating with the system.	
		3.	Overall System Operation		
			a. Operation of each circuit shall be demonstrated.	Ten hours every month.	
			b. Volume flow rate through charcoal filter shall be shown to be between 4500 and 12,000 cfm.	~~At least once per plant operating cycle.~~ R	
		4.	Manual initiation of the system shall be demonstrated.	~~At least once per plant operating cycle.~~ R	

**Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

TABLE 3-5
(Continued)

	Test		Frequency	USAR ~~FSAR~~ Section Reference
10c. Charcoal Adsorbers for S.I. Pump Room	1.	In-Place Testing** Charcoal adsorbers shall be leak tested and shall show ≥ 99% Freon (R-11 or R-112) removal.	On a refueling frequency ~~Each refueling shutdown not to exceed 18 months~~ or ~~after~~ every 720 hours of system operation or after each complete or partial replacement of the charcoal adsorber bank, or after any major structural maintenance on the system housing or ~~and~~ following significant painting, fire or chemical release in any ventilation zone communicating with the system.	9.10 6.2
	2.	Laboratory Testing a. Initial batch tests of all charcoal adsorbers shall show ≥ 99% elemental iodine removal when tested under conditions of ≥ 95% R.H., ≥ 125°F, 5 to 10 mg/m^3 inlet elemental iodine concentration and at a face velocity within ± 20% of system design.	Prior to initial loading in the filter unit.	
		b. The carbon sample test results for the S.I. Pump Room charcoal filters shall show no less than 90% elemental iodine removal, under conditions of ≥ 95% R.H., at ≥ 125°F, 5 to 10 mg/m^3 inlet elemental iodine concentration and within ± 20% of design face velocity.	On a refueling frequency ~~Each refueling shutdown not to exceed 18 months~~ or ~~after~~ every 720 hours of system operation or ~~and~~ following significant painting fire or chemical release in any ventilation zone communicating with the system.	
	3.	Overall System Operation a. Operation of each circuit shall be demonstrated.	Ten hours every month.	
		b. Volume flow rate shall be shown to be between 3000 and 6000 cfm.	~~At least once per plant operating cycle.~~ R	

** Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

TABLE 3-5 (Continued)

				Test	Frequency	USAR Section Reference
10c.	(continued)	4.	Automatic and/or manual initiation of the system shall be demonstrated.		~~At least once per plant operating cycle.~~ R	
11.	Containment Cooling and Iodine Removal Fusible Linked Dampers	1.	Demonstrate damper action.		1 year, 2 years, 5 years, and every 5 years thereafter.	9.10
		2.	Test a spare fusible link.			
12.	Diesel Generator Under-Voltage Relays		Calibrate		~~During each refueling outage.~~ R	8.4.3
13.	Motor Operated Safety Injection Loop Valve Motor Starters (HCV-311, 314, 317, 320, 327, 329, 331, 333, 312, 315, 318, 321)		Verify the contactor pickup value at $\leq$85% of 460 V.		~~During each refueling outage.~~ R	
14.	Pressurizer Heaters		Verify control circuits operation for post-accident heater use.		~~During each refueling outage.~~ R	
15.	Spent Fuel Pool Racks		Test neutron poison samples for dimensional change, hardness change, weight, neutron attenuation change and specific gravity change.		1, 2, 4, 7, and 10 years after installation, and every 5 years thereafter.	
16.	Reactor Coolant Gas Vent System	1.	Verify all manual isolation valves in each vent path are in the open position.		During each refueling outage just prior to plant start-up.	
		2.	Cycle each automatic valve in the vent path through at least one complete cycle of full travel from the control room. Verification of valve cycling may be determined by observation of position indicating lights.		~~During each refueling outage.~~ R	
		3.	Verify flow through the reactor coolant vent system vent paths.		~~During each refueling outage.~~ R	

Table 3-5
(Continued)

	Test	Frequency
17. Hydrogen Purge System[1]	1. Verify all manual valves are operable by completing at least one cycle.	R ~~During each refueling outage~~
	2. Cycle each automatic valve through at least one complete cycle of full travel from the control room. Verification of the valve cycling may be determined by the observation of position indicating lights.	R ~~During each refueling outage~~
	3. Initiate flow through the VA-80A and VA-80B blowers, HEPA filter, and charcoal adsorbers and verify that the system operates for at least (a) 30 minutes with suction from the auxiliary building (Room 59) (b) 10 hours with suction from the containment	a) ~~Monthly~~ M b) ~~During each refueling outage~~ R
	4. Verify the pressure drop across the VA-82 HEPAs and charcoal filter to be less than 6 inches of water. Verify a system flow rate of greater than 80 scfm and less than 230 scfm during system operation when tested in accordance with 3b. above.	R ~~During each refueling outage~~

~~(1) The requirements of T.S. requirement 2.9.1(2)f(iii) do not apply.~~

(6) Azimuthal Power Tilt (Tq)

Whenever the core power is above 70% of rated power, the azimuthal power tilt shall be determined to be within its limits by calculating the tilt at least once every day using either:

excore

a. The ~~encore~~ detectors with at least four safety channels operable, or

b. The incore detectors with at least two strings of three rhodium detectors per full core height quadrant operable.

(7) DNB Parameters

a. The cold leg temperature, pressurizer pressure, and axial shape index shall be verified to be within the limits of Section 2.10.4(5) at least once per shift.

b. The reactor vessel coolant total flow rate shall be determined to be within its limit by measurement at least once per month.

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 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 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 Chairperson of the Safety Audit and Review Committee.~~ DELETED

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

k. Review of the Fire Protection Program Plan and shall submit changes to the ~~Chairman~~ Chairperson of the Safety Audit and Review Committee.

l. 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.

5.5.2.8

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

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

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

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.

j. 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.

k. 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.

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 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 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.

5.8.2.1 Each procedure, or change thereto, shall be reviewed by a Qualified Reviewer (QR) who is knowledgeable in the functional area affected but is not the individual preparer. The QR may be from the same line-organization as the preparer. The QR shall render a determination in writing of whether or not cross-disciplinary review of a procedure, or change thereto is necessary. If necessary, such review shall be performed by appropriate personnel.

5.8.2.2 Each procedure, or change thereto, shall be reviewed by the Department Head designated by Administrative Controls Standing Orders as the responsible Department Head for that procedure, and the review shall include a determination of whether or not a 10 CFR 50.59 safety evaluation is required. If a 10 CFR 50.59 safety evaluation is not required, the procedure, or change thereto, shall be approved by the responsible Department Head or the Manager-Fort Calhoun Station, prior to implementation. Administrative Controls Standing Orders~~, the Site Security Plan and Implementing Procedures, the Emergency Plan and Implementing Procedures,~~ and the Fire Protection Program Plan shall be reviewed in accordance with Specification 5.5.1.6 and approved by the Manager-Fort Calhoun Station.

5.8.2.3 If the responsible Department Head determines that a procedure, or change thereto, requires a 10 CFR 50.59 safety evaluation, the responsible Department Head shall render a determination in writing of whether or not the procedure, or change thereto, involves an Unreviewed Safety Question (USQ) and shall forward the procedure, or change thereto with the associated safety evaluation to the PRC for review in accordance with Specification 5.5.1.6.a. If a USQ is involved, NRC approval is required prior to implementation of the procedure, or change.

5.8.2.4 Qualified Reviewers shall meet or exceed the respective qualifications for either Supervisors Requiring an AEC License, Professional-Technical Personnel, or Technical Support Personnel, as specified in ANSI N18.1 - 1971. Personnel recommended to be QRs shall be reviewed by the PRC and approved and designated as such by the PRC Chairman. The responsible Department Head shall ensure that a sufficient complement of QRs for their functional area is maintained in accordance with Administrative Controls Standing Orders.

5.8.2.5 Each procedure of Specification 5.8.1 shall be reviewed periodically as set forth in Administrative Controls Standing Orders.

5.8.2.6 Records documenting the activities performed under Specifications 5.8.2.1 through 5.8.2.4 shall be maintained in accordance with Specification 5.10.

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LIC-95-0005

ATTACHMENT B

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATIONS

DISCUSSION AND JUSTIFICATION:

In accordance with GL 93-07 (Reference 2), Omaha Public Power District (OPPD) is proposing administrative revisions to Specification 5.5.1.6 Items i & j, Specification 5.5.2.8 Items e, f & g and Specification 5.8.2.2. Revisions proposed for Specifications 5.5.1.6 Items i & j remove the review of the emergency and site security plans and implementing procedures from the list of responsibilities of the Plant Review Committee (PRC) and Safety Audit and Review Committee (SARC). Revisions proposed for Specification 5.5.2.8 Items e, f & g remove the audit of the emergency, site security and safeguards contingency plans and implementing procedures from the responsibilities of the SARC. The revision proposed for Specification 5.8.2.2 removes the review and approval of the emergency and site security plans and implementing procedures from the list of responsibilities of the Manager-Fort Calhoun Station.

As stated in GL 93-07, provisions sufficient to address these requirements are contained in 10 CFR Parts 50 and 73. OPPD utilizes the respective plan and/or administrative procedures to assure compliance with 10 CFR Parts 50 and 73. Therefore, it is unnecessary to restate these requirements in the Technical Specifications (TS). However, in accordance with GL 93-07, TS 5.5.1.6i and TS 5.8.2.2 requirements for PRC and SARC review and Manager-Fort Calhoun Station review and approval of the site security plan and implementing procedures will be fully incorporated into the site security plan during the requested implementation period. The emergency plan and implementing procedures already require PRC and SARC review and Manager-Fort Calhoun Station review and approval, in accordance with the provisions of GL 93-07.

Specification 2.5

The revision proposed for Specification 2.5 deletes unnecessary detail specifying that a backup water supply to the emergency feedwater storage tank from the Missouri River through the fire water system shall be available. Although backup water for the emergency feedwater storage tank will still be required, the proposed wording does not specify the source of the water since several other preferred sources of water are available. These sources include the water plant demineralized water system and the outside condensate storage tank (reference: Updated Safety Analysis Report (USAR), Section 9.4.6).

Specification 2.8

Specification 2.8(8) and statements in the bases of Specification 2.8 are proposed for deletion. Specification 2.8(8) requires a test of fuel handling cranes that will be required to handle spent fuel assemblies during refueling operations. Based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993 (58 FR 39132), it is not necessary to retain Specification 2.8(8) in the Fort Calhoun Station (FCS) Technical Specifications. Controls and limitations for the operation and testing of the fuel handling cranes will be incorporated into the USAR. The requirements of Specification 2.8(8) are currently contained in Station procedures to ensure that the handling of fuel assemblies and control element assemblies (CEAs) is accomplished safely and effectively.

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATIONS

DISCUSSION AND JUSTIFICATION (Continued):

This revision makes the FCS Technical Specifications more similar to NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," which does not contain requirements concerning the operation of fuel handling cranes.

Specification 2.11

Specification 2.11, which describes restrictions on the Containment Building and Auxiliary Building overhead cranes is proposed for deletion. Based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993, Specification 2.11 does not need to be retained in the FCS Technical Specifications. Controls and limitations for the operation and testing of the Containment Building and Auxiliary Building overhead cranes will be incorporated into the USAR.

The restrictions of Specification 2.11 are currently contained in Station procedures to ensure that the handling of loads over the reactor coolant system (RCS) and spent fuel storage pool is accomplished in accordance with the guidance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This revision makes the FCS Technical Specifications more similar to Standard Technical Specifications (STS), which do not contain restrictions on the operation of overhead cranes.

Specification 3.2

Revisions proposed for Specification 3.2 are as follows:

1. Specification 2.9.1(2)f(iii) was removed by Amendment 152. Therefore, references to Specification 2.9.1(2)f(iii) contained in the basis of TS 3.2 and the footnote to Item 17 of Table 3-5 are being deleted.

2. Wherever possible, text in Table 3-5 specifying the frequency of surveillance testing will be replaced with symbols defined in Specification 3.0.2. The frequency of testing associated with the symbols is equivalent to that specified in the text they replace.

3. It is proposed to revise Table 3-5, Item 1, to require testing CEA drop times prior to reactor criticality, after each removal of the reactor vessel closure head. Currently, Table 3-5, Item 1, states that the surveillance is to be performed at each refueling operation. The definition of refueling operation includes the shuffling of fuel in the spent fuel storage pool. Thus, a plant shutdown would be required to test CEA drop times whenever fuel is shuffled in the spent fuel storage pool, which is unjustified since this evolution does not affect the ability of the CEAs to drop into the core. The proposed frequency is the most appropriate time to perform the surveillance to ensure that the CEAs drop into the core within the time specified in the safety analysis, and is identical to the frequency of STS 3.1.5.7.

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATIONS

DISCUSSION AND JUSTIFICATION (Continued):

4. It is proposed to delete Table 3-5, Item 5, which requires testing the refueling system interlocks prior to the refueling outage. The wording of Item 5 is incorrect because it is not possible to test the interlocks on Fuel Handling Machine FH-1 prior to the refueling outage; the reactor vessel closure head must be removed before the interlocks can be tested. Therefore, based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993, Table 3-5, Item 5, may be deleted from the FCS Technical Specifications since these requirements are currently contained in Station procedures. Controls and limitations for the operation and testing of the refueling system interlocks will be incorporated into the USAR.

5. The frequency listed in Table 3-5, Item 10 is being revised by replacing occurrences of "Each refueling shutdown not to exceed 18 months" with "On a refueling frequency," which is defined in Specification 3.0.2 as "At least once per 18 months." This revision assures consistent use of terminology among the frequencies specified in Table 3-5. Secondly, concerning Item 10 of Table 3-5, the word "after" is being deleted from "or after every 720 hours of system operation." Removal of the word "after" introduces additional operational flexibility such that the surveillance could be performed before 720 hours of system operation are reached, if necessary. Finally, to clarify Item 10 of Table 3-5, the phrase "and following significant painting" is being changed to "or following significant painting."

6. Table 3-5 references to "FSAR" are being changed to "USAR" to reflect current terminology.

Specification 3.10

The revision proposed for Specification 3.10(6)a. corrects a misspelled word. The word "encore" is being revised to "excore."

Specification 5.5.1.6

The revision proposed for Specification 5.5.1.6, Item k revises the title of "Chairman" to "Chairperson" to be consistent with TS 5.5.2.2.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed revisions do not involve significant hazards considerations because operation of Fort Calhoun Station (FCS) in accordance with these revisions would not:

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

The proposed revisions to Technical Specifications (TS) 5.5 and 5.8 are administrative in nature and follow the guidance of Generic Letter (GL) 93-07. The review and audit functions of the site security and emergency plans and procedures will be retained in a manner that fully satisfies regulatory requirements. Therefore, the proposed revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to TS 2.5 will still require backup water for the emergency feedwater storage tank to be available. However, several other available sources of water are preferred over river water, such as, the water plant demineralized water system and the outside condensate storage tank. Therefore, the proposed revision does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed deletion of TS 2.8(8) pertaining to fuel handling cranes, deletion of TS 2.11 pertaining to overhead cranes in the Containment and Auxiliary Buildings, and deletion of statements in the bases of TS 2.8 pertaining to crane interlocks does not involve a significant increase in the probability or consequences of an accident previously evaluated. Specifications 2.8(8), 2.11 and the deleted statements in the bases of Specification 2.8 need not be retained in the TS based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993 (58 FR 39132).

Controls and limitations for the operation and testing of these cranes and interlocks will be incorporated into the Updated Safety Analysis Report (USAR). The requirements of TS 2.8(8) and restrictions of TS 2.11 are currently contained in Station procedures to ensure that the handling of fuel assemblies, control element assemblies (CEAs) and heavy loads is accomplished safely and effectively. These revisions make the FCS Technical Specifications more similar to Standard Technical Specifications (STS), which do not contain requirements or restrictions concerning the operation of fuel handling cranes or overhead cranes.

The revision proposed for TS 3.2, Table 3-5, Item 1 will make its surveillance frequency identical to the frequency specified in STS 3.1.5.7. The proposed frequency will require testing CEA drop times prior to reactor criticality after each removal of the reactor vessel closure head, which is the most appropriate time to perform the surveillance. The proposed frequency will ensure that the CEAs drop into the core within the time specified in the safety analysis and, therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed deletion of TS 3.2, Table 3-5, Item 5, which currently requires testing refueling system interlocks prior to the refueling outage does not involve a significant increase in the probability or consequences of an accident previously evaluated. Table 3-5, Item 5, does not need to be retained in the TS based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993. Controls and limitations for testing the refueling system interlocks will be incorporated into the USAR. The requirements for testing refueling system interlocks are already contained in Station procedures. This revision makes the FCS Technical Specifications more similar to STS, which do not contain requirements or restrictions pertaining to testing refueling system interlocks.

The proposed revision to TS 3.2, Table 3-5, Item 10, ensures consistent use of terminology among the frequencies specified in Table 3-5. The proposed revision clarifies the wording and introduces additional operational flexibility such that the surveillance could be performed before 720 hours of system operation, if warranted by plant conditions or beneficial to plant operation. Therefore, the proposed revision does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The remaining TS revisions are administrative in nature in that they correct references, titles, misspelling(s), and page numbers, or revise wording to be consistent with defined intervals within the TS. Therefore, they do not increase the probability or consequences of an accident previously evaluated. None of the proposed TS revisions will impact the function or method of operation of plant systems, structures, or components.

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

The proposed revisions to TS 5.5 and 5.8 which delete the review and/or audit of the emergency, site security and safeguards contingency plans and implementing procedures from the TS are administrative in nature and in accordance with the guidance of GL 93-07. The proposed revisions will not affect the operation of any system, structure, or component and therefore do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed revision to TS 2.5 will still require a backup supply of water for the emergency feedwater storage tank to be available. However, several other available sources of water are preferred over river water, such as, the water plant demineralized water system and the outside condensate storage tank. Therefore, the proposed revision does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed deletion of TS 2.8(8) pertaining to fuel handling cranes, deletion of TS 2.11 pertaining to overhead cranes in the Containment and Auxiliary Buildings and deletion of statements in the bases of TS 2.8 pertaining to crane interlocks does not create the possibility of a new or different kind of accident from any accident previously evaluated. Specifications 2.8(8), 2.11 and the deleted statements in the bases of Specification 2.8 need not be retained in the TS based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993.

The requirements of TS 2.8(8) and restrictions of TS 2.11 are currently contained in Station procedures to ensure that the handling of fuel assemblies, CEAs and heavy loads is accomplished safely and effectively. These revisions make the FCS Technical Specifications more similar to STS, which do not contain requirements or restrictions concerning the operation of fuel handling cranes or overhead cranes.

The proposed revision to TS 3.2, Table 3-5, Item 1, is an administrative revision to the frequency of CEA drop time testing. The proposed frequency is the most appropriate time to perform the surveillance to ensure that the CEAs drop into the core within the time specified in safety analysis and is identical to the frequency specified in STS 3.1.5.7. Therefore, the proposed revision does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed deletion of TS 3.2, Table 3-5, Item 5, which currently requires testing the refueling system interlocks prior to the refueling outage, does not create the possibility of a new or different kind of accident from any accident previously evaluated. Table 3-5, Item 5, does not need to be retained in the TS based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993. The requirements for testing refueling system interlocks are currently contained in Station procedures. This revision makes the FCS Technical Specifications more similar to STS, which do not contain requirements or restrictions pertaining to testing refueling system interlocks.

The proposed revision to TS 3.2, Table 3-5, Item 10, ensures consistent use of terminology among the frequencies specified in Table 3-5. The proposed revision clarifies the wording and introduces additional operational flexibility such that the surveillance could be performed before 720 hours of system operation, if warranted by plant conditions or beneficial to plant operation. Therefore, the proposed revision does not create the possibility of a new or different kind of accident from any previously evaluated.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION (Continued):

The remaining TS revisions are administrative in nature in that they correct references, titles, misspelling(s), and page numbers, or revise wording to be consistent with defined intervals within the TS. Therefore they do not create the possibility of a new or different kind of accident.

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

The proposed revisions to TS 5.5 and 5.8 concerning the review and/or audit of the emergency, site security and safeguards contingency plans and implementing procedures do not involve a significant reduction in a margin of safety. The audit and review processes are administrative functions which will be retained outside the TS in a manner that fully satisfies regulatory requirements.

Removing the requirement of TS 2.5 that Missouri River water from the fire water system shall be available to provide a backup water supply to the emergency feedwater storage tank improves operational flexibility without reducing any safety margins. Better sources of backup water are available to replenish the emergency feedwater storage tank. Although deleted from TS 2.5, the fire water system is still required to be available to meet the requirements of paragraph 3.F of the FCS Operating License. Therefore, the proposed revision does not involve a significant reduction in a margin of safety.

The proposed deletion of TS 2.8(8) pertaining to fuel handling cranes, deletion of TS 2.11 pertaining to overhead cranes in the Containment and Auxiliary Buildings and deletion of statements in the bases of TS 2.8 pertaining to crane interlocks does not involve a significant reduction in a margin of safety. Specifications 2.8(8), 2.11 and the deleted statements in the bases of Specification 2.8 do not need to be retained in the TS based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993.

The requirements of Specification 2.8(8) and restrictions of Specification 2.11 are currently contained in Station procedures to ensure that the handling of fuel assemblies, CEAs and heavy loads is accomplished safely and effectively. These revisions make the FCS Technical Specifications more similar to STS, which do not contain requirements or restrictions concerning the operation of fuel handling cranes or overhead cranes.

The proposed revision to TS 3.2, Table 3-5, Item 1, is an administrative revision to the frequency of CEA drop time testing. The proposed frequency is the most appropriate time to perform the surveillance to ensure that the CEAs drop into the core within the time specified in the safety analysis and is identical to the frequency specified in STS 3.1.5.7. Therefore, the proposed revision does not involve a significant reduction in a margin of safety.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION (Continued):

The proposed deletion of TS 3.2, Table 3-5, Item 5, which currently requires testing the refueling system interlocks prior to the refueling outage does not involve a significant reduction in a margin of safety. Table 3-5, Item 5, does not need to be retained in the TS based upon Criteria 1 through 4 of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 22, 1993. The requirements for testing refueling system interlocks are currently contained in Station procedures. This revision makes the FCS Technical Specifications more similar to STS, which do not contain requirements or restrictions pertaining to testing refueling system interlocks.

The proposed revision to TS 3.2, Table 3-5, Item 10, ensures consistent use of terminology among the frequencies specified in Table 3-5. The proposed revision clarifies the wording and introduces additional operational flexibility such that the surveillance could be performed before 720 hours of system operation if warranted by plant conditions or beneficial to plant operation. Therefore, the proposed revision does not involve a significant reduction in a margin of safety.

The remaining TS revisions are administrative in nature in that they correct references, titles, misspelling(s), and page numbers, or revise wording to be consistent with defined intervals within the TS. Therefore, they do not involve a significant reduction in a margin of safety.

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