2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

Objective

To prevent the principal process variables from exceeding a safety limit.

Specification

a. Reactor trip settings shall be as follows:

1. Nuclear Flux
    - A. Source Range (high set point) - within span of source range instrumentation
    - B. Intermediate range (high set point) $\leq$ 40 % of rated power 45
    - C. Power range (low set point) $\leq$ 25% of rated power
    - D. Power range (high set point) $\leq$ 109% of rated power
    - E. Power range fast flux rate trip (positive) 15% Δq/ 5 sec
    - F. Power range fast flux rate trip (negative) 10% Δq/ 5 sec
2. Pressurizer
    - A. High pressurizer pressure $\leq$ 2385 psig
    - B. Low pressurizer pressure $\geq$ 1875 psig
    - C. High pressurizer water level $\leq$ 90% of full scale
3. Reactor Coolant Temperature
    - A. Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2(T-T')\left[\frac{1 + \tau_1 s}{1 + \tau_2 s}\right] + K_3 (P-P') - f(\Delta I)\right]$

      where

      ΔT_o = Indicated ΔT at rated power, ^oF

      T = Average temperature, ^oF

      T' = 567.3 ^oF

      P = Pressurizer pressure, psig

      P' = 2235 psig

      K_1 = 1.11 24

      K_2 = 0.0090

      K_3 = 0.000566

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be operable whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power operated relief valves are set for test pressure plus 35 psi and the charging pump has a safety valve to protect the system.

B. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressurizer Heaters

A. At least one group of pressurizer heaters shall have an emergency power 45
supply available when the average RCS temperature is greater than 350^{o}F.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part 1 of the specification requires that both reactor coolant pumps be operating when the reactor is in power operation to provide core cooling in the event that a loss of flow occurs. Planned power operation with one loop out of service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this mode of operation. The flow provided in each case in Part 1 will keep DNBR well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted for any length of time except for tests. Upon loss of one pump below 10 percent full power the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation will remove decay heat up to 10 percent power. Above 10 percent power, an automatic reactor trip will occur if flow from either pump is lost.[1]

Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at set point. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against over-pressurization.

Pressurizer heaters are vital elements in the operation of the pressurizer which is necessary to maintain system pressure. Loss of energy to the heaters would result in the inability to maintain system pressure via heat addition to the pressurizer. This could result in an uncontrolled depressurization due to heat loss to ambient. Hot functional tests[2] have indicated that one group of heaters is required to overcome ambient heat losses. Placing heaters necessary to overcome ambient heat losses on emergency power will assure the ability to maintain pressurizer pressure. Annual surveillance tests are performed to ensure heater operability. 45

References:

(1) FSAR Section 7.2.2

(2) Hot functional test (PT-RC-31) 45

TABLE 4.1-3

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Equipment Tests***	Test	Frequency	Maximum Time Between Test (Days)	
1.	Control Rods	Rod drop times of all full length rods	Each refueling outage	N.A.	
		Partial movement of all rods	Every 2 weeks	17	
1a.	Reactor Trip Breakers	Open trip	Monthly	37	
1b.	Reactor Coolant Pump Breakers-Open-Reactor Trip	Operability	Each refueling outage	N.A.	
2.	Pressurizer Safety Valves	Set point	One each refueling outage	N.A.	
3.	Main Steam Safety Valves	Set point	Two each refueling outage	N.A.	
4.	Containment Isolation Trip	Operability	Each refueling outage	N.A.	
5.	Refueling System Interlocks	Operability	Prior to each refueling outage	N.A.	
6.	Ventilation System	Halide, DOP and Methyl Iodide Pressure Drop Test Visual Inspection	During each refueling outage except as specified in Note**	N.A.	
	a. Sheild Building				12
	b. Auxiliary Building SV Zone				
	c. Spent Fuel Pool				
7.	Fire Protection Pump and Power Supply	*Operability	Monthly	37	
8.	Containment Leak Detect	Operability	Weekly	8	
9.	Diesel Fuel Supply	*Fuel inventory	Weekly	8	
10.	Turbine Stop and Governor Valves	Operability	Monthly(1)	37(1)	
11.	Fuel Assemblies	Visual Inspection	Each refueling outage	N.A.	
12.	Guard Pipes	Visual Inspection	Each refueling outage	N.A.	
13.	Pressurizer Heaters	Operability****	Once each refueling cycle	N.A.	45

NOTES:

* See Specification 4.1.d

** Tests and frequency shall be in accordance with Specifications 4.4.d and 4.12.

*** Following maintenance on the above equipment that could affect the operation of the equipment tests should be performed to verify operability.

**** Test will verify operability of heaters and availability of an emergency power supply. 45

(1) This test may be waived for end of cycle operations when boron concentrations are less than 150 ppm, due to operational limitations.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

Specification

a. The reactor shall not be heated above 350°F unless the following conditions are satisfied:

   1. Rated relief capacity of TEN steam system safety valves is available. except during testing.
   2. TWO of the three auxiliary feedwater pumps are operable.
   3. System piping and valves directly associated with the above components are operable.
   4. A minimum of 75,000 gallons of water is available in the condensate storage tanks, except as specified in 3.4.a.4.a, and the Service Water System is capable of delivering an unlimited supply from Lake Michigan. 45
      a. Condensate storage tank level may drop below 75,000 gallons during recovery from a shutdown for a period not to exceed 12 hours.
   5. The iodine-131 activity on the secondary side of the steam generators does not exceed 1.0 μCi/cc.

b. If, when the reactor is above 350°F, any of the conditions of Specification 3.4.a cannot be met within 48 hours, the reactor shall be shutdown and cooled below 350°F using normal operating procedures.

Basis

A reactor shutdown from power requires removal of core decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

The ten main steam safety valves have a total combined rated capability of 7,765,000 lbs/hr. The maximum full-power steam flow is 7,449,000 lbs/hr; therefore, the ten main steam safety valves will be able to relieve the total steam flow if necessary.

In the unlikely event of complete loss of electrical power to the plant, continued capability of decay heat removal would be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump is normally aligned with one steam generator; the discharge of the turbine-driven pump, which starts automatically, is manually valved as necessary to backup either or both motor-driven pumps, or to replace the standby function of either motor-driven pump when it is out of service. Any single auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the reactor.

The specified minimum water supply in the condensate storage tanks is sufficient for two hours of hot shutdown plus cooling of the plant to the cold shutdown condition and maintenance of this condition for an additional four hours. During startup when water usage is at a maximum, the condensate storage tank level may drop below 75,000 gallons, since this mode of operation requires less water in 45 storage to assure sufficient cooldown capability (i.e., less decay heat). In the unlikely event of a need for more water an unlimited supply is available from Lake Michigan through the Service Water System. 45

The secondary coolant activity is based on a postulated release of the contents of one steam generator to the atmosphere. This could happen, for example, as a result of a steam break accident combined with failure of a steam line isolation valve. The limiting dose for this case results from iodine-131, because of its low MPC, and because its long half-life relative to the other iodine isotopes results in its greater concentration in the liquid. The accident is assumed to occur at zero load when the steam generators contain maximum water. With allowance for plate-out retention in water droplets, one-tenth of the contained iodine is assumed released from the plant. The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot \chi/Q \cdot DCF$$

where: C = secondary coolant activity, 1.0 μCi/cc

V = water volume in one steam generator, 3510 ft^3 = 99 m^3

$B(t)$ = breathing rate, 3.47×10^{-4} m^3/sec

χ/Q = 2.9×10^{-4} sec/m^3

DCF = 1.48×10^{6} rem/Ci iodine-131 inhaled

The resultant dose is less than 1,5 rem,

References:

FSAR Section 10

FSAR Section 14.1

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the engineered safety features.(1) Section 2.3 of these specifications describes the limiting safety settings for the protective instrumentation.

Safety Injection

The Safety Injection System is actuated automatically to provide emergency cooling and reduction of reactivity in the event of a loss-of-coolant accident or a steam line break accident.

Safety injection in response to a loss of coolant accident is initiated upon
receipt of low pressurizer pressure signals, and, in the case of rapid RCS 45
depressurization, by a high containment pressure signal.

Safety injection in response to a steam line break is provided directly by a low steam line pressure signal, backed up by the coincidence signal from the pressurizer and, in the case of a break within the containment, by the high containment pressure signal.

The Safety injection of highly borated water will offset the temperature-induced reactivity addition that could otherwise result from cooldown following a steam line break.

4.10 ENVIRONMENTAL MONITORING

Applicability

Applies to the periodic monitoring and recording of radioactive effluents and the routine testing of plant environs.

Objective

To verify that radioactive releases are maintained within allowable limits and that plant operations have no significant detrimental effects on the environment.

Specification

a. Environmental samples shall be collected and analyzed according to Table TS 4.10-1, where minor changes in descriptions of specific sampling locations may occur as necessitated by private ownership.

b. Reports shall be submitted in accordance with Section 6.9 of the Technical Specifications. 45

Basis

The operational program of environmental monitoring described in Section 2.8 of the FSAR will have been in progress for more than two years before initial plant startup. The number and distribution of sampling locations and the various types of measurement, together with the pre-operational background data, will provide verification of the effectiveness of plant effluent control and indication of measurable changes in the activity of the environment.

TABLE TS 4.10-1 (Page 1 of 6)

Operational Environmental Radiological Surveillance Program

Type of Sample	Location	Sampling Frequency	Type of Analysis	Frequency of Analysis	Reporting Units	Approximate Minimum Detectable Level	Comments
A. Airborne Particulates	K-1f K-2 K-7 K-8 K-15 K-16	Weekly	Gross alpha Gross beta Gamma Scan	Weekly Weekly Quarterly	pCi/m^3 pCi/m^3 pCi/m^3	4×10^{-4} pCi/m^3 1×10^{-3} pCi/m^3	On all samples On all samples Quarterly composite for each station
B. Airborne Iodine	Same as A	Bi-weekly	I-131	Bi-weekly	pCi/m^3	1×10^{-2} pCi/m^3	On all samples
C. Ambient Beta-Gamma	K-1f K-2 K-3 K-4 K-5 K-6 K-7 K-8 K-15 K-16						
TLD (5 chips in each packet)		Quarterly Annually	Beta-Gamma Beta-Gamma	Quarterly Annually	mrems/Q mrems/A	10 mrem	On all samples On all samples

3

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TABLE TS 4.10-1 (Page 5 of 6)

Operational Environmental Radiological Surveillance Program

Type of Sample	Location	Sampling Frequency	Type of Analysis	Frequency of Analysis	Reporting Units	Approximate Minimum Detectable Level	Comments
M. Bottom Sediments	500' North of discharge (on the beach) K-1d 500' South of discharge (on the beach) K-9 K-14	4 times per year	Gross alpha	4/year	pCi/g	**Same as Soil**	**May, July, Sept., Nov.**
			Gross beta	4/year	pCi/g		April or May, June August, and October
			Sr-89	4/year	pCi/g		April or May, June, August, and October
			Sr-90	4/year	pCi/g		April or May, June August, and October
N. Deleted							
O. Periphyton (Slime) and Aquatic Plants	K-1a K-1b K-1d K-1e K-9 K-14	Semi-annually	Gross alpha	Semi-annually	pCi/g	**0.11 pCi/g wet wt.**	2nd and 3rd quarters if available in sufficient quantity
			Gross beta	Semi-annually	pCi/g	**0.1 pCi/g wet wt.**	2nd and 3rd quarters if available in sufficient quantity
			Sr-89	Semi-annually	pCi/g	**0.01 pCi/g wet wt.**	2nd and 3rd quarters if available in sufficient quantity
			Sr-90	Semi-annually	pCi/g	**0.007 pCi/g wet wt.**	2nd and 3rd quarters if available in sufficient quantity

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Proposed Amendment No. 45

TABLE TS 4.10-2

(Page 1 of 1)
Sampling Locations

Code	Location
K-1	Onsite
1a	North Creek
1b	Middle Creek
1c	North of condenser
1d	Condenser discharge
1e	South Creek
1f	Meteorological tower
1g	South Well
1h	North Well
1i	WPS Emergency Center Well
K-2	WPS Operations Building in Kewaunee
K-3	Farm - 6.2 mi N of site
K-4	Farm - 3.0 mi N of site
K-5	Farm - 3.25 mi NNW of site
K-6	Farm - 6.7 mi WSW of site
K-7	Farm - 2.6 mi SSW of site
K-8	Catholic Church - 5.0 mi WSW of site
K-9	Rostok Water Intake for Green Bay, Wisconsin, two miles north of Kewaunee
K-10	Farm - 1.2 mi NNE of site
K-11	Farm - 1.0 mi NW of site
K-12	Farm - 1.0 mi W of site
K-13	General Store - 3 mi S of site
K-14	Two Creeks Park - 2.5 miles south of site
K-15	Gas Substation - 9 mi NNW of site
K-16	WPS Division Office Building, Green Bay, Wisconsin
K-17	Farm - 4.2 mi W of site
K-18	Food Stand - 7 mi SSW of site
K-19	Farm - 1.75 mi N of site

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