Docket No. 50-213
B13269

Attachment 1

Haddam Neck Plant

Proposed Revised Technical Specifications

June 1989

Section 2.0

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop inlet temperature (T_{cold}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop inlet temperature (T_{cold}) and pressurizer pressure has exceeded the appropriate percent of rated thermal power (RTP) line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to be within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

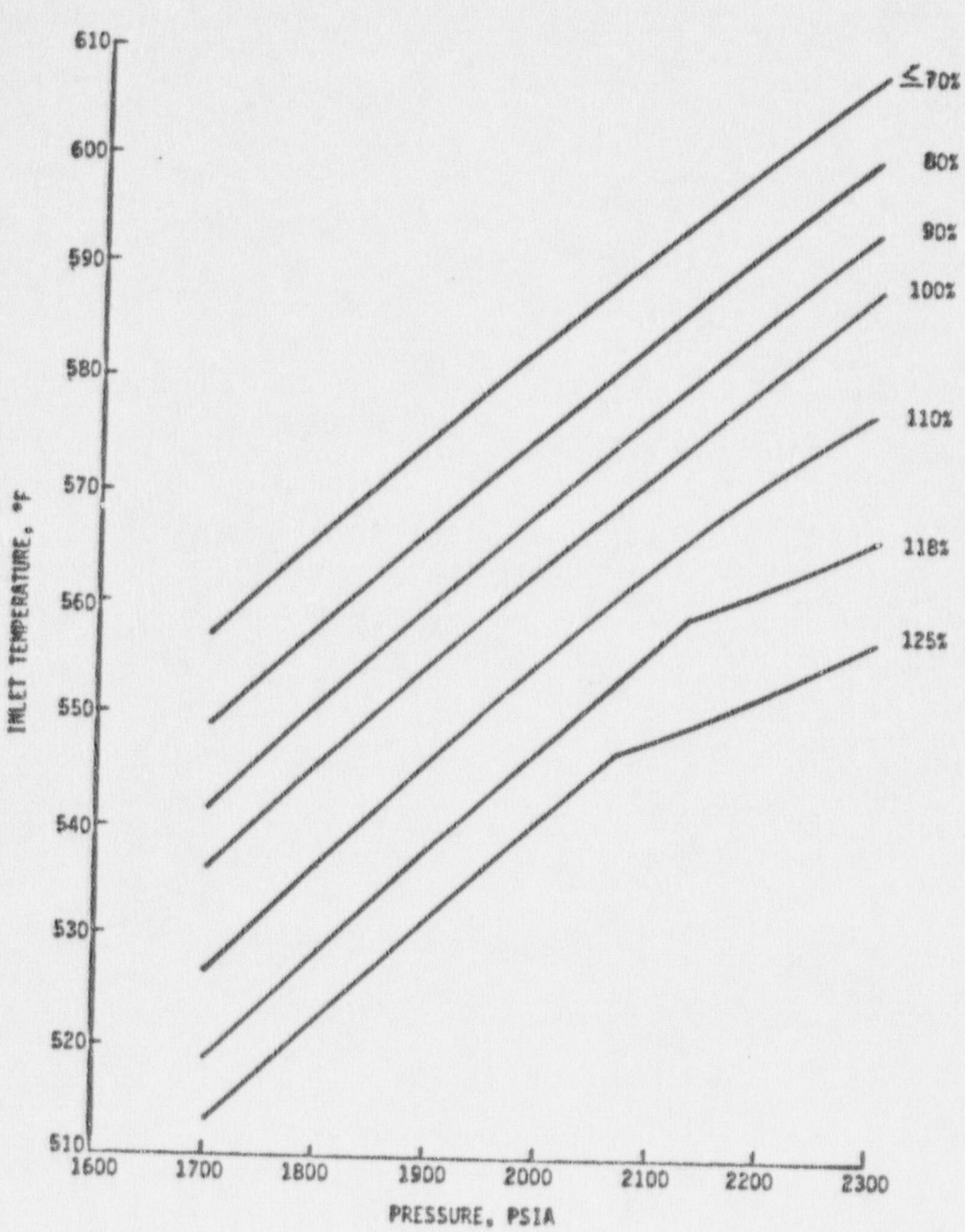

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

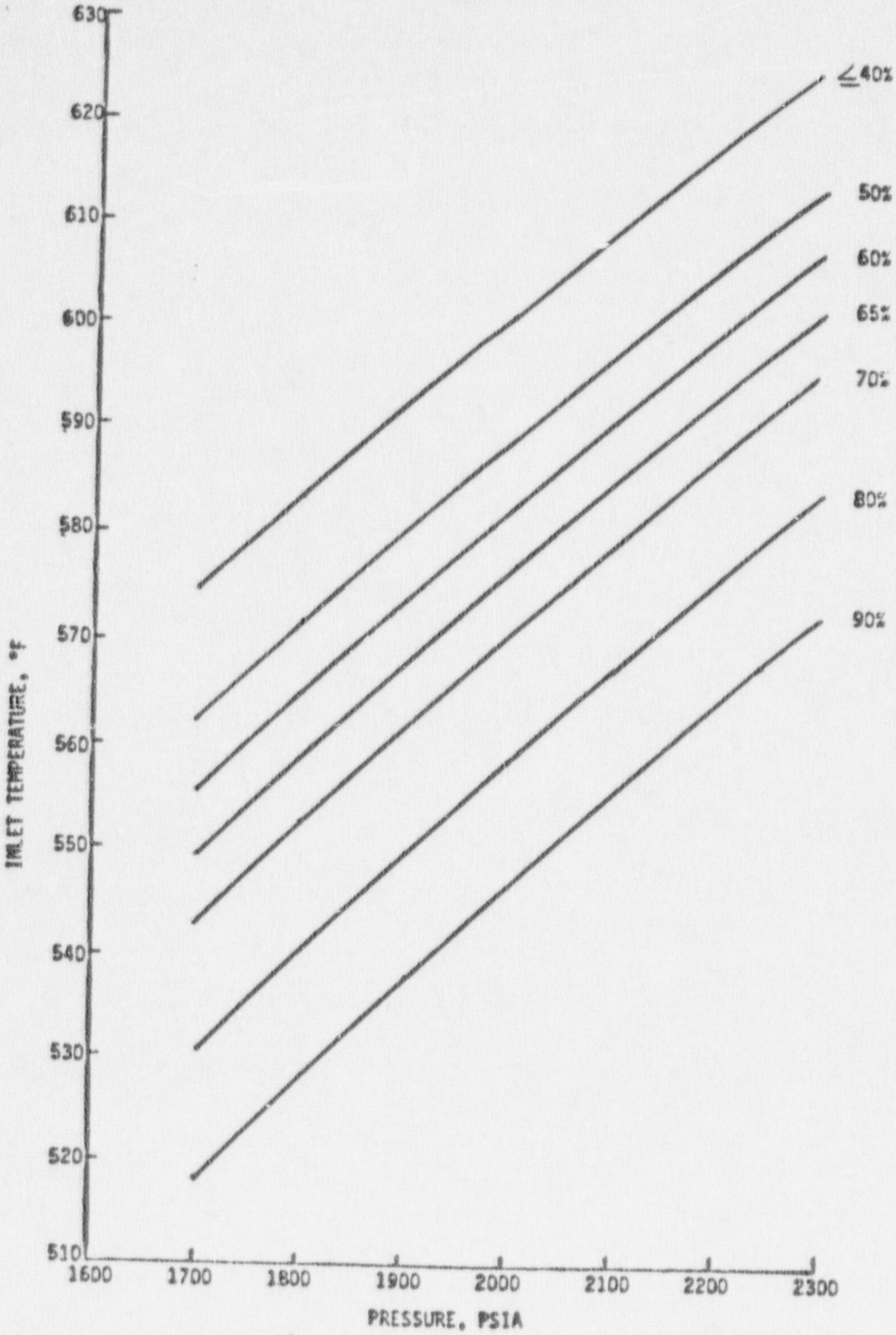

FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	$\leq$107% of RTP*	$\leq$109% of RTP*
b. Mid Setpoint	$\leq$72% of RTP*	$\leq$74% of RTP*
c. Low Setpoint	$\leq$23% of RTP*	$\leq$25% of RTP*
3. Intermediate Range, Neutron Flux, High Startup Rate	$\leq$4 DPM**	$\leq$5 DPM**
4. Pressurizer Pressure - Variable, Low		
a. Variable Setpoint	$\geq$17.4 (Tavg + 1.17 delta T) - 8835 psig	$\geq$17.4 (Tavg + 1.17 delta T) -8850 psig
b. Low Setpoint	$\geq$1720 psig	$\geq$1700 psig
5. Pressurizer Pressure--High	$\leq$2280 psig	$\leq$2300 psig
6. Pressurizer Water Level--High	$\leq$84% of instrument span	$\leq$86% of instrument span
7. Reactor Coolant Flow - Low (four loop and three loop operation)	$\geq$92% of nominal four loop flow per loop	$\geq$90% of nominal four loop flow per loop
	$\geq$86% of nominal three loop flow per loop	$\geq$84% of nominal three loop flow per loop

* RTP = RATED THERMAL POWER.
** DPM = Decades per minute.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Steam Flow - High	$\leq$108% of steam flow at RTP*	$\leq$110% of steam flow**** at RTP*
9. Steam Generator Water Level - Low Coincident With Steam/Feedwater Flow Mismatch	$\geq$13% of narrow range instrument span $\leq$19% of steam flow at RTP*	$\geq$11% of narrow range instrument span $\leq$20% of steam flow at RTP*
10. Undervoltage-Reactor Coolant Pumps	$\geq$70 volts on 120 volt bus	N.A.***
11. Safety Injection Input from ESF	N.A.	N.A.
12. Reactor Coolant Pump Breaker Position-Open	N.A.	N.A.***
13. Main Steam Line Trip Valve Closure	N.A.	N.A.
14. Turbine Trip	N.A.	N.A.***

* RTP - RATED THERMAL POWER.
*** This function not credited in the Safety Analyses.
**** Following a refueling outage, the calibration is performed subsequent to the plant reaching RTP.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
15. Reactor Trip System Interlocks		
a. Low Power Block, P-7		
1) Power Range, Neutron Flux	$\geq$7% and $\leq$8% of RTP*	$\geq$5% and $\leq$10% of RTP*
2) Turbine First Stage Pressure	$\geq$7% and $\leq$8% of RTP* at Turbine First Stage Pressure Equivalent	$\geq$5% and $\leq$10% of RTP* at Turbine First Stage Pressure Equivalent
b. Low Flow Permissive, P-8		
1) Power Range, Neutron Flux	$\leq$72% of RTP*	$\leq$74% of RTP*
2) Turbine First Stage Pressure	$\leq$72% of RTP* at Turbine First Stage Pressure Equivalent	$\leq$74% of RTP* at Turbine First Stage Pressure Equivalent
16. Reactor Trip System Breakers	N.A.	N.A.

* RTP = RATED THERMAL POWER

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value bounds a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, pressurizer pressure and core inlet temperature for which the minimum DNBR is no less than 1.30, and the core outlet void fraction is no greater than 0.32.

These curves are based on total enthalpy hot channel factors, F^N_H, of 1.60 and 1.64 for four and three loop operation, respectively. An allowance is included for an increase in F^N_H at reduced power.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. This insertion limit is described in Specification 3.1.3.6 and shown in Figures 3.1-1 and 3.1-2. The required reduction in power level as dictated by Figures 3.1-1 and 3.1-2 insures that the DNB ratio is always greater than 1.30.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel and pressurizer are designed to ASME Boiler and Pressure Vessel Code, Section VIII; ASME Special Ruling No. 1270N and 1273N, and the Tentative Structural Design Basis for Reactor Pressure Vessel and directly associated Components - PB151987. A maximum transient pressure of 110% (2735 psig) of design pressure is allowed by PB151987. The Reactor Coolant System piping, valves and fittings, are designed to ASA B 31.1 1955 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their acceptance criteria during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than a drift allowance accounted for in the design basis analysis.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

The Power Range Neutron Flux trip provides core protection against rapid reactivity excursions. In order to provide protection over the entire operating range, the trip function has three (3) different setpoints. The trip is credited in the following design basis events; steam line break, control rod ejection, excess steam flow, control rod withdrawal and isolated loop startup. In order to reduce the time to trip for certain accidents occuring at low power, the overpower setpoint is lowered to 23 percent when reactor power is below 10 percent. This low overpower is below 10 percent. This low overpower trip would terminate the postulated large steamline break accident from the hot zero power condition. The lower setting for three loop operation provides protection at the reduced power level equivalent to that provided by the setting for four loop operation at full power.

Intermediate and Source Range, Neutron Flux

The Intermediate Range, Neutron Flux, High Positive Rate trip provides core protection during reactor startup. This trip function provides protection for large reactivity insertion events initiated from a subcritical mode of operation. This trip function is credited in the rod withdrawal from subcritical analysis.

Pressurizer Pressure

The Variable Low Pressurizer Pressure trip protects the core against DNB or excessive core exit quality resulting from either uncontrolled slow reactivity insertions which cause Reactor Coolant System temperature and pressure to increase or a loss of RCS pressure. The formula for the Variable Low Pressure Trip Setpoint, which is based on reactor coolant temperature rise (Delta T) and Reactor Coolant System average temperature, defines a minimum allowable pressure for operation which is continually compared to pressurizer pressure. A Reactor trip occurs when the minimum allowable calculated pressure exceeds the pressurizer pressure. The Variable Low Pressurizer Pressure trip is credited in the uncontrolled rod withdrawal and steam generator tube rupture analyses.

The Pressurizer High Pressure trip, in conjunction with safety valves, protects the Reactor Coolant System against system overpressure. This trip is credited in the loss of load and turbine trip analyses.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer Code safety valves. This trip provides redundant protection to the Pressurizer High Pressure trip in the loss of load, and turbine trip analyses.

Reactor Coolant Flow

The Reactor Coolant Low Flow trip protects the core against DNB resulting from a reduction in coolant flow while the reactor is at substantial power. This trip is credited in the partial and total loss of flow, locked rotor, and sheared shaft analyses. Loss-of-flow protection is also provided by Reactor Coolant Pump Breaker trip and from Undervoltage trip on a reactor coolant pump motor bus. Credit was not taken in the design basis analyses for operation of the latter two trips but their functional capability enhances the overall reliability of the Reactor Trip System.

Steam Flow

The High Steam Flow Trip provides protection against a large increase in steam flow by closing the main steam line trip valves and tripping the reactor. This trip is credited in the steam line break and excess steam flow analyses.

Steam/Feedwater Flow Mismatch and Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Water Level - Low trip protects the RCS against an abrupt loss of secondary heat sinks by initiating a reactor trip prior to steam generator dryout. The trip is credited in the loss of feedwater analyses.

Undervoltage - Reactor Coolant Pump Buses

A Reactor trip is generated on low voltage on either reactor coolant pump bus. The trip provides protection for certain loss of flow events.

Safety Injection Input from ESF

The ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-2. This logic acts as a redundant trip to the Pressurizer Pressure low trip.

Reactor Coolant Pump Breaker

A Reactor trip from an opening of the reactor coolant pump breaker provides protection from loss of flow in any reactor coolant loop due to power failure.

Main Steam Line Trip Valve Closure

A Reactor trip on MSTV closure anticipates the pressure and flux transients which could follow MSTV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage, more than one reactor coolant pump breaker open, main steam line isolation valve closure, Turbine trip, and variable low pressure. On decreasing power, the above listed trips are automatically blocked.

P-7N On increasing power, P-7N automatically blocks the intermediate range, neutron flux, high startup rate trip. On decreasing power, P-7N automatically enables the intermediate range, neutron flux, high startup rate trip.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops, and one or more reactor coolant pump breakers open. On decreasing power, the P-8 automatically blocks the above listed trips.

Section 3/4.3

Instrumentation

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*,4*,5*	1 10
2. Power Range, Neutron Flux, a. High Setpoint b. Mid Setpoint c. Low Setpoint	4	2	3	1,2,3*,4*,5*	2#, 10
3. Intermediate Range, Neutron Flux, High Start Up Rate	2 2	1 1	1 2	2,3*,4*,5*** 5*	4 4
4. Pressurizer Pressure-Variable, Low	4	2	3	1(a)	6#
5. Pressurizer Pressure--High	3	2	2	1, 2	6#
6. Pressurizer Water Level--High	3	2	2	1(a)	6#
7. Reactor Coolant Flow - Low a. Above P-8	4 (1/loop)	1	(1/loop)	1(b)	7
b. Above P-7 and Below P-8	4 (1/loop)	2**	4 (1/loop)	1(c)	8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Steam Flow-High	4 (1/steam line)	2	1/steam line	1, 2	9
9. Steam Generator Water Level-Low Coincident With Steam/Feedwater Flow Mismatch	1/SG level and 1/steam/feed-water flow mismatch in each SG	1/SG level coincident with 1/steam/feed-water flow mismatch in same loop	1/SG level and 1/steam/feed-water flow mismatch in each SG	1, 2	5
10. Undervoltage - Reactor Coolant Pumps	2 (1/bus)	1	2 (1/bus)	1[(a)]	8
11. Safety Injection	2	1	2	1, 2	1

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	4 (1/pump)	1	4 (1 per pump)	1(b)	7
b. Above P-7 and Below P-8	4 (1/pump)	2**	4 (1 per pump)	1(c)	8
13. Steam Line Isolation Valve Closure	1/valve	1	1/valve	1(a)	8
14. Turbine Trip	3	2	2	1(a)	2
15. Reactor Trip System Interlocks					
a. Low Power Block, P-7					
1) Power Range, Neutron Flux	4	2	3	1(c)	3a
2) Turbine First Stage Pressure	1	1	1	1(c)	3b
b. Low Flow Permissive, P-8					
1) Power Range, Neutron Flux	4	2	3	1(b)	3a
2) Turbine First Stage Pressure	1	1	1	1(b)	3b
16. Reactor Trip System Breakers	2	1	2	1,2	11
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the Reactor Trip System breakers in the closed position and the Control Rod Drive System capable of rod withdrawal.

** The low flow channel associated with trip functions derived from the out-of-service reactor coolant loop shall be in the tripped condition.

*** With the Reactor Trip System breakers in the open position and the Control Rod Drive System not capable of rod withdrawal.

\# The provisions of Specification 3.0.4 are not applicable.

(a) THERMAL POWER above 10% of RATED THERMAL POWER.

(b) THERMAL POWER $\geq$ 74% of RATED THERMAL POWER.

(c) THERMAL POWER above 10% but below 74% of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2:

With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 2: (Continued)

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 3:

a. With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition or apply Specification 3.0.3.

b. With turbine first stage pressure inoperable, continued power operation may proceed provided the permissive is placed in the more conservative state for existing plant conditions.

ACTION 4:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes and restore the inoperable channel to OPERABLE status within 8 hours or open/verify open the Reactor Trip System breakers within the next hour.

ACTION 5:

a. With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the affected portion of the inoperable channel is placed in the tripped condition within 1 hour. The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 6:

With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours; however, the inoperable channel may be bypassed up to 8 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, reduce THERMAL POWER to below 74% of RATED THERMAL POWER (P-8) within 1 hour and place the inoperable channel in the trip position within the next 8 hours.

ACTION 8:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, reduce THERMAL POWER to below 10% of RATED THERMAL POWER (P-7) within 4 hours.

ACTION 9:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided that the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 10:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement for Modes 3, 4, 5, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 11:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement for Modes 3, 4, 5 be in at least HOT STANDBY within 6 hours.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	S/U	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	D(2,3)				
a. High Setpoint	S	R(3)	B	N.A.	1, 2
b. Mid Setpoint	S	R(3)	B	N.A.	1, 2
c. Low Setpoint	S	R(3)	B	N.A.	$1^{(d)}$, 2, 3*, 4*, 5*
3. Intermediate Range, Neutron Flux, High Startup Rate	S	R(3)	S/U(6)	N.A.	2, 3*, 4*, 5
4. Pressurizer Pressure--Variable, Low	S	R	SW	N.A.	$1^{(a)}$
5. Pressurizer Pressure--High	S	R	SW	N.A.	1, 2
6. Pressurizer Water Level--High	S	R	SW	N.A.	$1^{(a)}$
7. Reactor Coolant Flow--Low	S	R	(7)	N.A.	$1^{(a)}$

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Steam Flow--High	S	R(5)	R(5)	N.A.	1, 2
9. Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	S	R	R	N.A.	1, 2
10. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	R	1[(a)]
11. Safety Injection	N.A.	N.A.	N.A.	R	1, 2
12. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R	1[(a)]
13. Main Steam Line Trip Valve Closure	N.A.	N.A.	N.A.	R	1[(a)]
14. Turbine Trip	N. A.	N.A.	N.A.	R	1[(a)]

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
15. Reactor Trip System Interlocks					
a. Low Power Block, P-7	N.A.	R	R	N.A.	$1^{(c)}$
b. Low Flow Permissive, P-8	N.A.	R	R	N.A.	$1^{(b)}$
16. Reactor Trip System Breakers	N.A.	N.A.	N.A.	S/U(1, 4)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* With the Reactor Trip System breakers in the closed position and the Control Rod Drive System capable of rod withdrawal.

(a) THERMAL POWER above 10% of RATED THERMAL POWER.
(b) THERMAL POWER $\geq$ 74% of RATED THERMAL POWER.
(c) THERMAL POWER above 10% but below 74% of RATED THERMAL POWER.
(d) THERMAL POWER below 10% of RATED THERMAL POWER.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4. are not applicable for entry into MODES 1 or 2.

This requirement is not applicable when the Power Range Channels have had their gains adjusted to maintain the 9% trip margin for steady state conditions at power levels other than 16%, 65%, and 100% RATED THERMAL POWER. When this exception is used, a heat balance calculation will continue to be performed on a daily basis to determine core power, and the power range channels will be verified daily to be 9% below the selected overpower trip setpoint.

(3) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(4) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip System breakers.

(5) Following a refueling outage, the calibration is performed subsequent to the plant reaching RTP. The provisions of Specification 4.0.4 are not applicable.

(6) If not performed in previous 7 days.

(7) Each scheduled shutdown if not tested or calibrated in preceding 6 months.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlock shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS instrumentation channel or interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-3, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-2 until the channel is restored to OPERABLE status with the Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock logic shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO ACTUATE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Start Diesel Generators, Containment Isolation).					
a. Manual Initiation	2	1	2	1, 2, 3	23
b. Containment Pressure--High	6(3/train)	2 in any one train	4(2/train)	1, 2, 3	20
c. Pressurizer Pressure--Low	3	2	2	1, 2, 3**	24*
2. Steam Line Isolation					
a. Steam Flow in Two Steam Lines--High				1, 2, 3	
1) Four Loops Operating	1/steam line	1 in each of any 2 steam lines	1/steam line		21*
2) Three Loops Operating	1/operating steam line	1***/any operating steam line	1/operating steam line		22

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO ACTUATE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Auxiliary Feedwater					
a. Wide Range Stm. Gen. Water Level--Low					
1) Four Loops Operating	8	2 Channels in any one train	8	$1^{(a)(b)}$	21*
2) Three Loops Operating	6	2 channels in any one train	6	$1^{(a)(b)}$	26
b. Trip of All Main Feedwater Pumps	1/pump	1 from each pump	1/pump	$1^{(a)}$	26
4. Emergency Bus Undervoltage					
a. 4.16 kV Bus Under-voltage-Level 1	3/bus	2/bus	2/bus	1, 2, 3, 4	24*
b. 4.16 kV Bus Undervoltage-Level 2	3/bus	2/bus	2/bus	1, 2, 3, 4	24*
c. 4.16 kV Bus Undervoltage-Level 3	3/bus	2/bus	2/bus	1, 2	24*

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO ACTUATE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Containment Isolation (Containment Air Recirculation System, Feedwater Isolation, Safety Injection)					
a. Containment Pressure-High	6(3/Train)	2 in any one train	4(2/Train)	1, 2, 3, 4	23
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-2 (Continued)

TABLE NOTATIONS

*The provisions of Specification 3.0.4 are not applicable.

**Trip function may be bypassed in this MODE when RCS pressure is less than 1800 psig.

***The channel(s) associated with the protective functions derived from the out-of-service reactor coolant loop shall be placed in the tripped mode.

(a) THERMAL POWER above 10% of RATED THERMAL POWER.

(b) For Surveillance Testing purposes, the train being tested will be placed in "Defeat" function.

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 22 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 25 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or reduce the THERMAL POWER to below 10% of RATED THERMAL POWER within the following 1 hour.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Start Diesel Generators, Containment Isolation).		
a. Manual Initiation	N.A.	N.A.
b. Containment Pressure--High	$\leq$4.7 psig	$\leq$5.0 psig
c. Pressurizer Pressure--Low	$\geq$1720 psig	$\geq$1700 psig
2. Steam Line Isolation		
a. Steam Flow in Two Steam Lines--High	$\leq$108% of full* steam flow.	$\leq$110% of full* steam flow.
3. Auxiliary Feedwater		
a. Steam Generator Water Level--Low	$\geq$47% of wide range instrument span.	$\geq$45% of wide range instrument span.
b. Trip of All Main Feedwater Pumps	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Emergency Bus Undervoltage		
a. 4.16 kV Bus Undervoltage - Level 1	≥2870** volts	≥2784** volts
b. 4.16 kV Bus Undervoltage - Level 2	≥3684 volts with a 9 second time delay.	≥3664 volts with a 8 second to 10 second time delay.
c. 4.16 kV Bus Undervoltage - Level 3	≥4019 volts with a 9 second time delay.	≥3999 volts with a 8 second to 10 second time delay.
5. Containment Isolation (Containment Air Recirculation System, Feedwater Isolation, Safety Injection)		
a. Containment Pressure - High	≤4.7 psig	≤5.0 psig
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and allowable values.	

TABLE NOTATIONS

* Rated Thermal Power

** Setpoint is by tap position. Time delay of device is inverse function of voltage. Device must change state within 0.95 - 1.05 seconds when the input voltage to the device goes from normal to zero volts instantaneously.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Start Diesel Generators, Containment Isolation)					
a. Manual Initiation	N.A.	N.A.	N.A.	R	1, 2, 3
b. Containment Pressure-High	D	R	R	N.A.	1, 2, 3
c. Pressurizer Pressure-Low	S	R	SW	N.A.	1, 2, 3
2. Steam Line Isolation					
a. Steam Flow in Two Steam Lines-High	S	R	R	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Auxiliary Feedwater					
a. Steam Generator Water Level-Low	S	R	M	N.A.	1
b. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	R	1
4. Emergency Bus Undervoltage					
a. 4.16 kV Bus Undervoltage - Level 1	N.A.	R	N.A.	M	1, 2, 3, 4
b. 4.16 kV Bus Undervoltage - Level 2	N.A.	R	N.A.	M	1, 2, 3, 4
c. 4.16 kV Bus Undervoltage - Level 3	N.A.	R	N.A.	M	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Containment Isolation (Containment Air Recirculation System, Feedwater Isolation, Safety Injection)					
a. Containment Pressure --High	D	R	R	N.A.	1, 2, 3, 4
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.				

3/4 3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-4 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

a. With no radiation monitoring channels for plant operations OPERABLE, take the ACTION shown in Table 3.3-4.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. RCS Leakage Detection					
1) Gaseous Radio-activity (R-12)	N.A.	1	1, 2, 3, 4	N.A.	30

ACTION STATEMENTS

ACTION 30 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. RCS Leakage Detection				
1) Gaseous Radioactivity (R-12)	S	R	M	1, 2, 3, 4

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detector System shall be OPERABLE with:

a. At least 11 of 13 northwest quadrant detector thimbles or at least 10 of 13 northwest quadrant detector thimbles if the inoperable locations are M8, M12, and N8.

b. At least one set of four quadrant symmetric thimbles, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detector System is used for:

a. Recalibration of the Excore Neutron Flux Detector System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and the LINEAR HEAT GENERATION RATE.

ACTION:

a. With the Moveable Incore Detector System inoperable due to less than the minimum required number of detector thimbles as required in 3.3.3.2a or b, penalty factors shall be applied when measuring $F^N_{\Delta H}$ LINEAR HEAT GENERATION RATE or QUADRANT POWER TILT RATIO; or during recalibration of the Movable Incore Detector System, as appropriate.

b. With the Movable Incore Detector System inoperable, due to insufficient movable detectors, drives or readout equipment, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detector System shall be demonstrated OPERABLE by verifying an acceptable voltage plateau for the incore detector(s) at least once per 24 hours when used as required by the above Applicability requirements.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring system instrumentation shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the seismic monitoring system inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 The above seismic monitoring system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 The above required seismic monitoring system actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. If it is determined that the magnitude of the event exceeded the Operating Basis Earthquake, then a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

TABLE 3.3-5

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Servo Accelerometer (SSA-302) Basemat-Cable Vault	0 to 0.59	1
2. Digital Cassette Accelerograph (DCA-300)**	± 5 Volts	1
3. Response Spectrum Analyzer (RSA-50)**	± 5 Volts	1
4. Playback System (SMR-102)**	± 5 Volts	1
5. Seismic Warning Panel (SWP-300)**	N/A	1

**All located in the Control Room

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Servo Accelerometer (SSA-302) Basemat-Cable Vault	M	R	SA
2. Digital Cassette Accelerograph (DCA-300)**	M	R	SA
3. Response Spectrum Analyzer (RSA-50)**	M	R	SA
4. Playback System (SMR-102)**	M	R	SA
5. Seismic Warning Panel (SWP-300)**	M	R	SA

**All located in the Control Room

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

TABLE 3.3-6

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT/LOCATION	MINIMUM OPERABLE
1. Wind Speed	
a. Baseline Elev. 33′	1
b. Nominal Elev. 200′	1
2. Wind Direction	
a. Baseline Elev. 33′	1
b. Nominal Elev. 196′	1
3. Air Temperature	
a. Baseline Elev. 33′	1
4. Delta T*	
a. Nominal Elev. 120′	1
b. Nominal Elev. 200′	1

* Delta T is the air temperature of the nominal elevation minus the air temperature at the 33′ baseline elevation.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT/LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Baseline Elev. 33′	D	SA
b. Nominal Elev. 200′	D	SA
2. Wind Direction		
a. Baseline Elev. 33′	D	SA
b. Nominal Elev.196′	D	SA
3. Air Temperature		
a. Baseline Elev. 33′	D	SA
4. Delta T*		
a. Nominal Elev. 120′	D	SA
b. Nominal Elev. 200′	D	SA

* Delta T is the air temperature of the nominal elevation minus the air temperature at the 33′ baseline elevation.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The accident monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-7.

ACTION:

a. As shown in Table 3.3-7.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-6.

TABLE 3.3-7

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Containment Pressure	2	1	1,2,3	38, 43
2.	Reactor Coolant Cold Leg Temperature - Wide Range	1/loop	1/loop	1,2,3	38
3.	Reactor Coolant Hot Leg Temperature - Wide Range	1/loop	1/loop	1,2,3	38
4.	Reactor Coolant Pressure - Wide Range	2	1	1,2,3,4	35, 36
5.	Pressurizer Water Level	3	2	1,2,3	42
6.	Steam Generator Pressure	1/steam generator	1/steam generator	1,2,3	36
7.	Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator	1,2,3	36
8.	Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	1,2,3	35, 36
9.	Refueling Water Storage Tank Water Level	2	1	1,2,3	35, 36
10.	Boric Acid Tank Solution Level	1	1	1,2,3	43
11.	Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator	1,2,3	42
12.	Reactor Coolant System Subcooling Margin Monitor	2	1	1,2,3	38, 40
13.	PORV Block Valve Position Indicator	1/Valve	1/Valve	1,2,3	41
14.	Safety Valve and PORV (Acoustic Flow Monitor)	1	1	1,2,3	41
15.	Containment Water Level-Wide Range	2	1	1,2,3	38, 43
16.	Containment Water Level-Narrow Range (Containment Sump Monitor)	1	1	1,2,3,4	45

TABLE 3.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
17.	Core Exit Thermocouples	16/core* (4/quadrant)	2/quadrant	1,2,3,	38, 43
18.	Main Stack-Wide Range Noble Gas Monitor	1	1	1,2,3,4***	37
19.	Containment Atmosphere-High Range Radiation Monitor	2	2	1,2,3,4	44
20.	Reactor Vessel Water Level	2**	1**	1,2,3	38, 39

TABLE NOTATIONS

* Quadrant IV has 3/quadrant.

** A channel is composed of eight sensors in a probe. A channel is OPERABLE if four or more sensors, i.e., one or more in the head region (upper two) and three or more in the plenum region (lower six), are OPERABLE.

*** During periods of high steam generator blowdown, the main stack wide range noble gas monitor may be isolated for the duration of blowdown. In these cases, the monitor must be returned to service for at least 3 hours at the end of each six day period to demonstrate operability.

TABLE 3.3-7 (Continued)

ACTION STATEMENTS

ACTION 35 - With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-7, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 36 - With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-7, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirements, return one channel to operable status within 7 days, or else prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining: the cause of the malfunction, the plans for restoring the channel to OPERABLE status, and a preplanned alternative method for estimating stack release rates during the interim.

ACTION 38 - With the number of OPERABLE channels less than the Total Number of Channels shown in Table 3.3-7, either restore the inoperable channel(s) to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status and alternate methods in effect for estimating the applicable parameter in the interim.

ACTION 39 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, either restore the inoperable channels(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

a. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

b. Restore the system to OPERABLE status at the next scheduled refueling.

TABLE 3.3-7 (Continued)

ACTION STATEMENTS

ACTION 40 - With the number of channels OPERABLE less than the MINIMUM CHANNELS OPERABLE, determine the subcooling margin once per 12 hours. Restore the system to OPERABLE status at the next scheduled refueling.

ACTION 41 - With any individual valve position indicator inoperable (a block valve position indicator or the acoustic flow monitor), obtain quench tank temperature, level and pressure information and monitor discharge pipe temperature once per shift to determine valve position. For the case of an inoperable block valve position indicator this action is not required if the PORV block valve is closed with power removed in accordance with Specification 3.4.4.

ACTION 42 - With the number of OPERABLE channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-7, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

ACTION 43 - With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-7, restore the inoperable channel(s) to OPERABLE status within 48 hours, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in effect for estimating the applicable parameter during the interim.

ACTION 44 - With less than the minimum channel(s) operable, restore the inoperable channel(s) to operable status within 48 hours or else establish alternate means to determine if significant fuel failure exists. If still inoperable after 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining: the cause of the inoperability, the plans for restoring operability, and the alternate means established.

ACTION 45 - With the number of channels operable less than the minimum channels OPERABLE requirement of Table 3.3-7, restore the inoperable channel(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 4.3-6

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment Pressure	D	R	1,2,3
2. Reactor Coolant Cold Leg Temperature - Wide Range	M	R	1,2,3
3. Reactor Coolant Hot Leg Temperature - Wide Range	M	R	1,2,3
4. Reactor Coolant Pressure - Wide Range	D	R	1,2,3,4
5. Pressurizer Water Level	M	R	1,2,3
6. Steam Generator Pressure	M	R	1,2,3
7. Steam Generator Water Level - Narrow Range	M	R	1,2,3
8. Steam Generator Water Level - Wide Range	M	R	1,2,3
9. Refueling Water Storage Tank Water Level	M**	R	1,2,3
10. Boric Acid Mix Tank Solution Level	W	R	1,2,3
11. Auxiliary Feedwater Flow Rate	M	R	1,2,3
12. Reactor Coolant System Subcooling Margin Monitor	M	R	1,2,3
13. PORV Block Valve Position Indicator	M	R	1,2,3
14. Safety Valve and PORV Acoustic Flow Monitor	M	R	1,2,3
15. Containment Water Level-Wide Range	M	R	1,2,3
16. Containment Water Level-Narrow Range (Containment Sump Monitor)	D	R	1,2,3,4

TABLE 4.3-6 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	MODES FOR WHICH SURVEILLANCE IS REQUIRED
17. Core Exit Thermocouples	M	R***	1,2,3
18. Main Stack-Wide Range Noble Gas Monitor	D	Q	1,2,3,4
19. Containment Atmosphere-High Range Radiation Monitor	D	R*	1,2,3,4
20. Reactor Vessel Water Level	M	R***	1,2,3

* CHANNEL CALIBRATION may consist of an electronic calibration of the channel not including the detector, for range decades above 10R/hr and a one point calibration of the detector below 10R/hr with an installed or portable gamma source.

** In addition to monthly CHANNEL CHECK, a quarterly functional test must be performed.

*** Electronic calibration from the ICC cabinets only.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: Whenever systems, structures, components, or equipment protected by the fire detection instrumentation are required to be OPERABLE.

ACTION:

a. With any, but no less than the minimum required fire detection instruments shown in Table 3.3-8 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).

b. With less than the minimum required fire detection instruments in any fire zone shown in Table 3.3-8 operable, within 1 hour establish a continuous fire watch to inspect the zone(s) with the inoperable instrument(s), unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months. Detectors which cannot be reset are not required to be demonstrated OPERABLE by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST.

SURVEILLANCE REQUIREMENTS

4.3.3.6.2 Supervised circuits associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.6.3 Nonsupervised circuits, associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.

TABLE 3.3-8

FIRE DETECTION SYSTEMS

Location	Minimum Number Smoke Detectors OPERABLE/Detectors Available	Minimum Number Heat Detectors OPERABLE/Detectors Available
1. Containment (R-3)	19/22 *	24/32 ** (Inaccessible)
2. Cable Spreading Area (S-3A)	21/28 *	
3. 1A Diesel Generator Room (D-1)	4/5	
4. 1B Diesel Generator Room (D-2)	4/5	
5. Switchgear Room (S-2)	35/35	
6. Containment Cable Vault (R-1)	3/4	
7. Waste Disposal Bldg. (W-I)	2/3	
8. Auxiliary Feedwater Pump Room (R-2)	1/2	
9. Primary Auxiliary Bldg.		
Entrance to Corridor, West End (A-1A)	1/2	
Main Corridor and Boric Acid Area (A-1A)	3/4	
East End of Corridor (A-1A)	1/1	
Northeast End of Corridor (A-1A)	1/1	
Drumming Room (A-1I)	2/3	
Ventilation Equipment Area (A-1N)	2/3	
Store Room (A-1P)	1/2	
RHR Pump Room Cubicles A and B (A-1E, A-1F)	1/1	
10. Control Room (S-1A, Not including kitchen)	13/18*	
11. Screen Well Bldg.		
Pump Motor Room (P-1A, P-1B)	8/9***	
Hypoclorite Storage Room (P-1C)	1/1	
12. Spent Fuel Bldg. (F-1)	5/6	
13. PAB Charcoal Filter Bank Heat Detector (A-1N) (outlet detectors only)		7/7

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*No two adjacent detectors shall be inoperable at the same time.
**Minimum of 6 heat detectors per reactor coolant pump must be OPERABLE.
***One detector inoperable cannot be one of the three existing detectors on the upper level of the Screen Well Building.

TABLE 3.3-8

FIRE DETECTION SYSTEMS

Location	Minimum Number Smoke Detectors OPERABLE/Detectors Available	Minimum Number Heat Detectors OPERABLE/Detectors Available
14. High pressure turbine deluge (T-1F)	-	4/4
15. Hydrogen seal oil reservoir deluge (T-1D)	-	1/1
16. Turbine lube oil reservoir deluge (T-1B)	-	7/7
17. Turbine building mezzanine under generator (T-1F)	-	4/4
18. Turbine building cranewell deluge (T-1C)	-	6/6

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with applicable Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints shall be determined in accordance with methodology and parameters described in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times*.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the Alarm/Trip Setpoint so it is acceptably conservative.

b. With the number of channels less than the minimum channels operable requirement, take the ACTION shown in Table 3.3-9. Exert best efforts to restore the inoperable monitor to OPERABLE status within 30 days, and, if unsuccessful, explain in the next Semi-annual Effluent Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-7.

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* At all times means that channel shall be OPERABLE and in service on a continuous, uninterrupted basis, except that outages are permitted for a maximum of 12 hours each time for the purpose of maintenance and performance of required tests, checks, calibrations or sampling.

TABLE 3.3-9

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM # OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Waste and Recycle Test Tank Discharge Line	1	40
b. Steam Generator Blowdown Line*	1	41
2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water Effluent Line	1	42
3. FLOW RATE MEASUREMENT		
a. Waste and Recycle Test Tank Discharge Line	1	43
b. Steam Generator Blowdown Line	**	N.A.
c. Discharge Canal	***	N.A.

* Automatic termination of blowdown requirement will become effective upon completion of proposed modification to provide automatic termination.

** Flow is determined by the use of valve curves for the purpose of determining flows only.

*** Discharge canal flow is determined by the use of pump curves.

TABLE 3.3-9 (Continued)

ACTION STATEMENTS

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, effluent releases may continue provided that best efforts are made to repair the instrument and that prior to initiating a release:

a. At least two independent samples of the tank to be discharged are analyzed in accordance with Specification 4.11.1.1.1, and;

b. The original release rate calculations and discharge valving are independently verified by a second individual.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 3×10^{-7} microcuries/ml;

1. Once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gm DOSE EQUIVALENT I-131.

2. Once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gm DOSE EQUIVALENT I-131.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that once per 12 hours grab samples of the service water effluent are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 3×10^{-7} microcuries/ml.

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that the flow rate is estimated once per 4 hours during actual releases. Pump performance curves generated insitu may be used to estimate flow.

TABLE 4.3-7

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Waste and Recycle Test Tank Discharge Line	D(1)	P	R(2)	Q(3)
b. Steam Generator Blowdown Line*	D(1)	M	R(2)	Q(3)
2. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water Effluent Line	D(1)	M	R(2)	Q(3)
3. FLOW RATE MEASUREMENT				
a. Waste and Recycle Test Tank Discharge Line	D(1)	N.A.	R	N.A.
B. Steam Generator Blowdown Line	D(5)	N.A.	N.A.	N.A.
c. Discharge Canal	D(4)	N.A.	N.A.	N.A.

* Automatic termination of blowdown requirement will become effective upon completion of a proposed modification to provide automatic termination.

TABLE 4.3-7 (Continued)

TABLE NOTATION

(1) CHANNEL CHECK need only be performed daily when discharges are made from this pathway. The CHANNEL CHECK should be done when the discharge is in process.

(2) CHANNEL CALIBRATION shall be performed using a known radioactive liquid or solid source whose strength is determined by a detector which has been calibrated to an NBS source. The radioactive source shall be in a known, reproducible geometry.

(3) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint*.

2. Instrument indicates a downscale failure or circuit failure.

3. Instrument controls not set in operate mode.

(4) Pump status should be checked at least once per 24 hours for the purpose of determining flow rate.

(5) Blowdown throttle valve position should be checked daily when discharges are being made via this pathway.

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* Automatic isolation shall also be demonstrated annually for the test tank discharge monitor line and steam generator blowdown line.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE with applicable Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the methodology and parameters as described in the ODCM.

APPLICABILITY: At all times*.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above Specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the Alarm/Trip Setpoint so it is acceptably conservative.

b. With the number of channels less than the minimum channels operable requirement, take the ACTION shown in Table 3.3-10. Exert best efforts to restore the inoperable monitor to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semi-annual Effluent Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

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* At all times means that the channel shall be OPERABLE and in service on a continuous basis, except that outages are permitted for a maximum of 12 hours each time for the purpose of maintenance and performance of required tests, checks, calibrations.

TABLE 3.3-10

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. MAIN STACK		
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Waste Gas System Releases	1	45
b. Iodine Sampler	1	46
c. Particulate Sampler	1	46
d. Stack Flow Rate Monitor	1	47
e. Sampler Flow Rate Monitor	1	47

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases via the Waste Gas Holdup System may continue provided that best efforts are made to repair the instrument and that prior to initiating the release:

(a) For the tank to be discharged, at least two independent samples of the tank's contents are analyzed; and,

(b) The release rate calculations and discharge valve lineups are independently verified by a second individual.

Otherwise, suspend releases from the Waste Gas Holdup System.

Releases from all pathways other than the Waste Gas Holdup System may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross radioactivity within 24 hours.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that samples are continuously collected with auxiliary sampling equipment for periods of seven (7) days and analyzed for principal gamma emitters with half lives greater than 8 days within 48 hours after the end of the sampling period. Auxiliary sampling shall be established within 12 hours of declaring the channel INOPERABLE.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that the best efforts are made to repair the instrument and that the flow rate is estimated once per 4 hours.

TABLE 4.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. MAIN STACK				
a. Noble Gas Activity Monitor	D(1)	M	R(2)	Q(3)
b. Iodine Sampler	W	N.A.	N.A.	N.A.
c. Particulate Sampler	W	N.A.	N.A.	N.A.
d. Stack Flow Rate Monitor	D(1)	N.A.	R	N.A.
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) CHANNEL CHECK daily when releases exist via this pathway.

(2) Calibration shall be performed using a known source whose strength is determined by a detector which has been calibrated to an NBS source. These sources shall be in a known, reproducible geometry.

(3) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint*.

b. Instrument indicates a downscale failure or circuit failure.

c. Instrument controls not set in operate mode.

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* Automatic isolation of the waste gas releases by the noble gas activity monitor should also be demonstrated.

3/4.3.4 INTERNAL FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 The liquid level instrumentation channels for flooding protection shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With less than the Minimum Channels OPERABLE for any liquid level instrumentation Functional Unit, take the ACTION shown in Table 3.3-11.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.4 Each liquid level instrumentation channel for flood protection shall be demonstrated OPERABLE:

a. At least once per 12 hours by performance of the CHANNEL CHECK,

b. At least once per 6 months by performance of an ANALOG CHANNEL OPERATIONAL TEST and visually verifying no obstruction.

TABLE 3.3-11

LIQUID LEVEL INSTRUMENTATION FOR FLOODING PROTECTION

FUNCTIONAL UNIT (Floor Elevation)	TO ALARM	MINIMUM CHANNELS OPERABLE	ALARM/FLOAT ACTION
1. RHR Pump Cubicle (-19′)	1	1	48
2. PAB East (21′6")	1	1	48
3. PAB West (21′6")	1	1	48
4. Pipe Chase East (13′3")	1	1	48
5. Pipe Chase West (14′2")	1	1	48
6. Metering Pump Cubicle (15′6")	1	1	48
7. Charging Pump A Cubicle (15′6")	1	1	48
8. Charging Pump B Cubicle (15′6")	1	1	48
9. Safety Injection Pump Cubicle (15′6")	1	1	49
10. Condensate Return Pump Cubicle (8′6") (Safety Injection Pump Area)	1	1	48
11. Drumming Room Sump (8′6")	1	1	48
12. Screenwell House (8′)	1	2	50
13. Diesel Generator B Room (21′6")	1	2	50
14. Diesel Generator A Room (21′6")	1	2	50

TABLE 3.3-11 (Continued)

ACTION STATEMENTS

ACTION 48 - With no channels OPERABLE, restore an inoperable channel(s) to OPERABLE status within 8 hours or within the next 1 hour establish a liquid level watch patrol to inspect the zone(s) without an OPERABLE channel at least once per hour.

ACTION 49 - With no channels OPERABLE for Functional Unit 9, Safety Injection Pump Cubicle, the Minimum Channels OPERABLE requirement for Functional Unit 10, Condensate Return Pump Cubicle, must be met or take the action specified in ACTION 48, above.

ACTION 50 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days or within the next 8 hours establish a liquid level watch patrol to inspect the zone(s) with the inoperable channel at least once per hour. With no channels OPERABLE, take the action specified in ACTION 48, above.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-3 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-3. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed,

REACTOR TRIP SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) startup of the emergency diesel generators, (4) containment isolation, (5) Turbine trip, (6) auxiliary feedwater pumps start and automatic valves position, (7) containment cooling fans start and automatic valves position, and (8) essential service water pumps start and automatic valves position.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the Containment Atmosphere Gaseous Radioactivity Monitoring System ensures that Gaseous Radioactivity Monitoring System will monitor inside containment as a means to detect RCS leakage.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.6 FIRE DETECTION INSTRUMENTATION

The OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility Fire Protection Program.

Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant's Fire Protection Program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.7 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the REMODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 FLOODING PROTECTION

The liquid level instrumentation is provided to monitor liquid levels in areas of potential flooding caused by local pipe ruptures. The system ensures that early warning will occur so that protective action can be taken in the event of a localized flooding condition in areas of the plant that house safety-related equipment. The loss of detection capability represents a degradation of flooding protection for any area. As a result, the establishment of a liquid level watch patrol must be initiated at an early stage. The establishment of frequent liquid level watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

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Attachment 2

Haddam Neck Plant

Description of Individual Proposed Changes to the Technical Specifications and Discussion on the Significant Hazards Consideration

Section 2.0 - Safety Limits and Limiting Safety System Settings

Section 3/4.3 - Instrumentation

June 1989

Docket No. 50-213
B13269

Attachment 2

Haddam Neck Plant

Technical Specification

Section 2.0, Safety Limits and
Limiting Safety System Settings

June 1989

Technical Specification Section 2.0
Safety Limits and Limiting Safety System Settings

Section 2.1, Safety Limits

The proposed revised Technical Specification (RTS) Section 2.1, Safety Limits, has been prepared by converting the existing Technical Specification Section 2.1, Introduction, Section 2.2 Safety Limits - Reactor Core, and Section 2.3, Safety Limits - Reactor Coolant System Pressure to a format consistent with the Westinghouse Standard Technical Specifications (W STS). The content of the proposed RTS sections is the same as the existing Technical Specifications. No changes have occurred to these sections other than the renumbering to achieve consistency with the W STS. The proposed RTS is an enhancement to the existing Technical Specifications. It provides clear applicability, action, and surveillance requirements modeled after the W STS. The proposed changes are compared to the existing Technical Specifications and the W STS. A matrix summarizing this comparison is included in Attachment 3.

Significant Hazards Consideration

In accordance with 10CFR50.92, CYAPCO has reviewed the proposed changes and has concluded that they do not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The content of the RTS is the same as the previously approved version of the existing Technical Specification Section 3.17. No changes have occurred to this section other than renumbering to achieve consistency with the W STS. Therefore, there is no increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes do not impact the operation of any component or system. The proposed changes do not introduce any new failures. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from those previously analyzed.

3. Involve a significant reduction in a margin of safety. Since the proposed changes do not affect the consequences of an accident previously analyzed, there is no reduction in the margin of safety.

Section 2.2, Limiting Safety System Settings

The proposed revised Technical Specification Section 2.2, Limiting Safety System Settings, has been prepared by converting the existing Technical Specification Section 2.4, Protective Instrumentation to a format consistent with the W STS. For additional discussion, refer to Section 3/4.3, Instrumentation.

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Attachment 2

Haddam Neck Plant

Technical Specification
Section 3/4.3 Instrumentation

June 1989

Technical Specification Section 3/4.3, Instrumentation

The proposed revised Technical Specification (RTS) Section 3/4.3 has been prepared by converting the existing Technical Specification Section 2.4, Protective Instrumentation, Section 3.8, Turbine Cycle, Section 3.9, Operational Safety Instrumentation and Control Systems, Section 3.11, Containment, Section 3.21, Safety-Related Equipment Flood Protection, Section 3.22 Fire Protection Systems Section, 3.23, Post Accident Instrumentation, Section 4.2, Operational Safety Items, Section 4.3, Core Cooling System - Periodic Testing, Section 4.8, Auxiliary Steam Generator Feed Pump, Section 4.14, Flood Protection Annunciation, Section 4.15 Fire Protection Systems, and Section 7/8.2, Instrumentation, to a format consistent with the Westinghouse Standard Technical Specifications (W STS). In addition, applicable Administrative Technical Specifications[1] at the Haddam Neck Plant have also been included in the proposed RTS. The proposed changes are compared to the existing Technical Specifications and the W STS. A matrix summarizing this comparison is included in Attachment 3.

Section 2.2.1 - Reactor Trip System Instrumentation and Section 3/4.3.1 - Reactor Trip System Instrumentation

The above two sections are tied closely together, therefore they are discussed under a common heading. The reactor trip setpoint limits specified in Table 2.2-1 have been selected to ensure that the reactor core and reactor coolant system (RCS) are prevented from exceeding their acceptance criteria during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System (ESFAS) in mitigating the consequences of accidents. The proposed RTS sections 2.2.1 and 3/4.3.1 have been prepared by converting existing Technical Specification Section 2.4, 3.9 and 4.2 to a format consistent with the W STS. The following is a description of the changes between the existing Technical Specifications and the proposed RTS. New and additional requirements are a conservative change and do not require further justification. Clarifications are considered to be a more detailed definition of the requirement or action and do not require justification. All relaxed or deleted requirements are justified below.

New/Additional Requirements

1. The trip settings given in the existing Technical Specification Section 2.4 corresponds to the allowable values in Table 2.2-1 of the proposed RTS. The proposed RTS also gives the actual trip setpoints, which are not in the existing Technical Specifications.

2. The applicability section for the proposed RTS Section 3.3.1 specifies mode requirements rather than just requiring them to be operable at full power as specified in the existing Technical Specifications.

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(1) Administrative Technical Specifications at the Haddam Neck Plant are administrative procedures that were implemented as an interim measure prior to converting the Technical Specifications to the W STS format.

3. The action for the proposed RTS Section 2.2.1 requires that channels that are out of calibration be declared inoperable.

4. The following reactor trip system instrumentation given in Table 2.2-1, Table 3.3-1 and Table 4.3-1 do not have any corresponding requirements in the existing Technical Specifications:

   - Item 10 - Undervoltage - Reactor Coolant Pump
   - Item 11 - Safety Injection
   - Item 12 - Reactor Coolant Pump Breaker Position Trip
   - Item 13 - Main Steam Line Trip Valve Closure (Except for Table 3.3-1)
   - Item 14 - Turbine Trip
   - Item 15 - Reactor Trip System Interlocks
   - Item 16 - Reactor Trip System Breakers

5. The high steam flow reactor trip requirements given as Item 8 of Table 4.3-1 do not have any corresponding requirements in the existing Technical Specifications.

6. The steam generator low level coincident with steam/feedwater mismatch trip requirements given as Item 9 of the proposed Table 2.2-1 do not have any corresponding requirements in the existing Technical Specifications.

7. Item 1 of the proposed Table 3.3-1 requires two manual trip channels instead of the one channel required by the existing Technical Specifications.

8. The requirements for a manual reactor trip are also included in the proposed RTS Table 2.2-1 and the surveillance requirements on the trip are added to the proposed Table 4.3-1.

9. Item 2 of the proposed Table 3.3-1 only allows 6 hours with 3 operable high power (neutron flux) trip channels. The existing Technical Specification allows indefinite operation under these conditions.

10. Item 9 of the proposed Table 3.3-1 requires an inoperable channel for low steam generator level coincident with steam/feedwater mismatch to be placed in the tripped condition within one hour. The existing Technical Specification allows operation to continue with continuous operator surveillance.

11. The following surveillance requirements are added to the proposed Table 4.3-1. They do not have any corresponding requirements in the existing Technical Specifications.

    - Item 2 - The channel calibration at refueling interval was added to the power range neutron flux instrumentation.

    - Item 3 - The channel calibration at the refueling interval was added to the intermediate range neutron flux instrumentation.

    - Item 4 - The analog channel operational test at a six week interval was added for the variable low pressure trip.

- Item 9 - The analog channel operational test was added at the refueling interval for the low steam generator level coincident with steam/feedwater mismatch trip.

Relaxed/Deleted Requirements

1. A footnote to Item 2 of the existing Technical Specification Section 2.4 allows the high pressurizer level trip to be bypassed when the reactor is at least 1.5% delta k subcritical. This may be interpreted to imply that it must be available in Modes 1, 2, and part of 3. The proposed RTS only requires the trip to be operable in Mode 1 above 10% power. The reduction in the mode requirement is acceptable since this trip is not credited in any safety analysis below 10% power.

2. A footnote to Item 8 of the existing Technical Specification Section 2.4 allows the high startup rate trip to be bypassed above 10% of rated power. This may be interpreted to imply that it is required below 10% power. The proposed RTS only requires the trip below 5% power. This relaxation is acceptable since the high startup rate trip is not required between 5% and 10% power for any design basis accidents.

3. Item 6 of the existing Technical Specification Section 2.4 gives the requirements for the reactor coolant loop valve temperature interlocks. These requirements are included in the proposed RTS surveillance 4.4.1.6.2. There is no corresponding limiting safety system settings or limiting condition for operation in the proposed RTS. The reactor coolant loop valve temperature interlock is considered a control grade rather than a safety grade. Therefore it is inappropriate for this interlock to appear in the proposed Table 2.2-1. The ACTION statement of the proposed RTS Section 3.4.1.6 duplicates the existing requirements of the specification 2.4.

4. The existing Technical Specification Section 2.4 gives a description of the bases of the various trip functions. Appropriate corresponding descriptions are moved to the bases of the proposed RTS.

5. Item C of the existing Technical Specification 3.9 gives specific operability requirements for neutron monitoring equipment. All reactor safety requirements are included in the proposed RTS Sections 2.2.1 and 3.3.1. Any other requirements are related to plant reliability and are not appropriate for inclusion in the Technical Specifications. Therefore they are not included in the proposed RTS.

6. The existing Technical Specification Table 3.9-1 requirement for the source range start-up rate rod stop is not included in the proposed RTS. This requirement is not credited in any of the plant's design basis safety analysis.

7. The requirements in the existing Technical Specification Table 3.9-1 for the shutdown high neutron level alarm is being deleted from the proposed RTS. Instead it will be included in the proposed RTS 3/4.9.2. This

section was submitted to the NRC[2] and will be revised accordingly and submitted to the NRC at a later date.

Section 3/4.3.2 - Engineered Safety Features Actuation System Instrumentation

The proposed RTS Section 3.3.2 has been prepared by converting the existing Technical Specification Sections 3.8, 3.9, 3.11, 4.2, and 4.8 to a format consistent with the W STS. The following is a description of the changes between the existing Technical Specifications and the proposed RTS. New and additional requirements are a conservative change and do not require further justification. Clarifications are considered to be a more detailed definition of the requirement or action and do not require justification. All relaxed or deleted requirements are justified below.

New/Additional Requirements

1. The following items are added to the proposed Table 3.3-2 and these items do not have any corresponding requirements in the existing Technical Specifications.

   - Item 1(a) - Manual Initiation of Safety Injection
   - Item 4 - Emergency Bus Undervoltage
   - Item 5 - Containment Isolation

2. Item 1(b) of the proposed Table 3.3-2 requires two trains of high containment pressure signals to be used for safety injection. The existing Technical Specifications only require one train.

3. Item 2(a)(2) of the proposed Table 3.3-2 requires one out of three logic for high steam flow steam line isolation. The existing Technical Specifications allow two out of three logic.

4. The trip setpoints from the proposed Table 3.3-3 do not have any corresponding items in the existing Technical Specifications for the following:

   - Item 1 - Safety Injection
   - Item 2 - Steam Line Isolation
   - Item 3 - Auxiliary Feedwater
   - Item 4 - Emergency Bus Undervoltage

5. Item 1(a) of the proposed Table 4.3-2 (manual initiation of safety injection) is not required in the existing Technical Specifications.

6. Item 4 (Emergency Bus Undervoltage) and Item 2 (Steam Line Isolation) included in the proposed Table 4.3-2 do not have any corresponding requirements in the existing Technical Specifications.

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(2) E. J. Mroczka letter to U.S. NRC, Revised Technical Specifications, dated October 26, 1988.

7. The following surveillance requirements are included in the proposed Table 4.3-2. There are no corresponding requirements in the existing Technical Specifications.

   - Item 3(a) - A shift channel check and refueling channel calibration for auxiliary feedwater initiation on low steam generator level.

   - Item 1(b) - A refueling analog channel operational check for safety injection on high containment pressure.

   - Item 5(a) - A refueling analog channel operational check for containment isolation on high containment pressure.

Clarifications

1. The action statement of Item 1(b) of the proposed Table 3.3-2 allows six hours in two out of two logic prior to shutdown. The existing Technical Specification does not give an allowable time frame to execute the action statement and applies for one operable train.

2. The action statement for Item 1(c) of the proposed Table 3.3-2 allows one hour in two out of two logic before placing an inoperable channel in trip. The existing Technical Specification does not specify a time frame.

3. The action statement for Item 2(a)(1) of the proposed Table 3.3-2 requires an inoperable channel to be placed in the trip mode in 1 hour. The proposed RTS allows one out of three operation at 100% power. The existing Technical Specification allows two out of three operation, but only with three loop operation.

4. The action statement for Item 3(a)(1) of the proposed Table 3.3-2 allows one hour to place an inoperable channel in trip for four loop auxiliary feedwater initiation on low steam generator level. The existing Technical Specification Table 3.8-1 does not specify the time requirements.

5. The action statement for Items 3(a)(2) of the proposed Table 3.3-2 allows 48 hours to fix an inoperable channel or the power must be reduced below 10%. The plant may run in a two out of two mode in one train during this time. The existing Technical Specification requires placement of an inoperable channel in the tripped mode, but does not require reduction in power.

Relaxed/Deleted Requirements

1. The requirements for manual initiation of auxiliary feedwater in Item a of the existing Technical Specification Table 3.8-1 are not included in Item 3 of the proposed Table 3.3-2. The requirement of manual initiation of the auxiliary feedwater is functionally met by manually starting the auxiliary feedwater pumps and opening the bypass valve. The existing auxiliary feedwater system does not have a separate manual initiation.

2. The action statement for Item 3(b) of the proposed Table 3.3-2 allows 48 hours to restore an inoperable auxiliary feedwater initiation channel or power must be reduced below 10%. The existing Technical Specification Table 3.8-1 does not give a time frame but does require the plant to remain in Modes 3 or 4. This change is acceptable since the decay heat loads below 10% power are small and would allow more than adequate time for the operator to manually initiate the auxiliary feedwater pumps. Therefore, this reduction in the action statement requirements is acceptable.

3. Item 14 (Residual Heat Pump Flow Instrumentation) of the existing Technical Specification Table 4.2-1 is not included in the RTS. This exclusion does not have a significant impact on plant safety since it is not credited in the design basis analysis. In addition, the W STS format does not require this parameter to be included in Technical Specifications.

Section 3/4.3.3.1 - Radiation Monitoring for Plant Operations

The proposed RTS provides limiting conditions of operation (LCO), action statements, surveillance requirements, and associated bases for radiation monitoring instrumentation during plant operation. The existing Technical Specifications do not address radiation monitoring instrumentation operability except for an operability test (Item 19 of Table 4.2-1). This proposed RTS clarifies that testing requirement by making it clear that the only radiation monitor with safety significance is the containment atmosphere gaseous radioactivity monitor and is consistent with the recently issued[3] amendment. The safety significance of this monitor is its capability to detect primary system leakage as required by the proposed RTS Section 3.4.6.1. This proposed RTS will help ensure the operability of this monitor for that purpose.

Section 3/4.3.3.2 - Movable Incore Detectors

The proposed RTS section provides LCOs, action statements, surveillance requirements and corresponding bases regarding movable incore detectors. The operability of the movable incore detectors with the specified minimum complement of equipment ensures that measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The existing Technical Specifications do not address the movable incore detectors operability.

Section 3/4.3.3.3 - Seismic Instrumentation

The proposed RTS section provides LCOs, action statements, surveillance requirements and corresponding bases regarding seismic instrumentation. The operability of the seismic instrumentation ensures that sufficient capability is available to determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required

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(3) A. B. Wang letter to E. J. Mroczka, Issuance of Amendment #116, dated May 31, 1989.

to permit comparison of the measured response to that used in the design basis for the facility. The existing Technical Specifications do not address the seismic instruments.

Section 3/4.3.3.4 - Meteorological Instrumentation

The proposed RTS section provides LCOs, action statements, surveillance requirements, and corresponding bases regarding meteorological instrumentation. The operability of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. The existing Technical Specifications do not address the meteorological instrumentation.

Section 3/4.3.3.5 - Accident Monitoring Instrumentation

The proposed RTS Section 3.3.3.5 has been prepared by converting the existing Technical Specification sections 3.9 and 3.23 to a format consistent with the W STS. The following is a description of the changes between the existing Technical Specifications and the proposed RTS. New and additional requirements are a conservative change and do not require further justification. The proposed Table 3.3-7 is equivalent to the W STS in most cases. However, in many cases, a number of action statements are less restrictive than those of the W STS but the proposed RTS is consistent with Generic Letter 83-37 and the recently issued amendment.(4) Applicability modes are either consistent with or more restrictive than those of the W STS.

New/Additional Requirements

1. The following items on the proposed Table 3.3-7 are required for Modes 1 through 3. These are only required in the existing Technical Specifications when the reactor is critical.

   - Item 5 - Pressurizer Water Level
   - Item 11 - Auxiliary Feedwater Flow Rate
   - Item 13 - PORV Block Valve Position Indicator
   - Item 14 - Safety Valve and PORV Acoustic Flow Monitor

2. The following items are added to the proposed Table 3.3-7. They do not have any corresponding requirements in the existing Technical Specifications.

   - Item 6 - Steam Generator Pressure
   - Item 7 - Narrow Range Steam Generator Water Level
   - Item 8 - Wide Range Steam Generator Water Level
   - Item 9 - Refueling Water Storage Tank Level
   - Item 10 - Boric Acid Solution Tank Level

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(4) A. B. Wang letter to E. J. Mroczka, Issuance of Amendment #113, dated April 24, 1989.

3. The following items on the proposed Table 4.3-6 have a monthly check required that is not required in the existing Technical Specifications.

   - Item 9 - Refueling Water Storage Tank Level
   - Item 13 - PORV Block Valve Position Indication

4. The following items are added to the proposed Table 4.3-6. They do not have any corresponding requirements in the existing Technical Specifications.

   - Item 6 - Steam Generator Pressure
   - Item 8 - Wide Range Steam Generator Water Level

Section 3/4.3.3.6 - Fire Detection Instrumentation

This proposed RTS section provides a Limiting Condition for Operation (LCO) requirement for the minimum number of OPERABLE fire detectors whenever systems, structures, components or equipment protected by the fire detection instrumentation are required to be OPERABLE. These requirements are equivalent to the existing Technical Specification with the following exceptions:

1) Instrumentation in additional fire zones are now included in the specification. This is an enhancement of the existing specification.

2) Additional requirements have been added when detectors have failed but the number operable still meet the minimum requirements. This is an enhancement of the existing specification.

3) A continuous fire watch is now required instead of a fire watch patrol when the number of operable detectors is not met. This is an enhancement of the existing specification.

4) Surveillance requirements are now included for nonsupervised circuits. This is an enhancement of the existing specification.

5) Surveillance requirements for detectors which cannot be reset have been deleted. These devices have links which melt to alarm and would have to be replaced each time they were tested if they were demonstrated operable by a Trip Actuation Device Operational Test. However, the circuits for these devices are still tested consistent with the remaining detector circuits to assure circuit integrity. This relaxation of surveillance requirements has been shown not to degrade the overall system reliability and is considered to provide equivalent protection.

6) The requirement for submitting a Special Report has been deleted. Each potential reportable event will be reviewed in accordance with the requirements of 10CFR50.73 as stated in proposed RTS Section 6.6.1.

3/4.3.3.7 - Radioactive Liquid Effluent Monitoring Instrumentation

The proposed RTS Section 3.3.3.7 has been prepared by converting the existing Technical Specification Section 7/8.2.1 to a format consistent with the W STS. The proposed RTS 3.3.3.7 is the same as the existing Technical Specification Section 7/8.2.1 except for an additional footnote added on the applicability statement. This footnote clarifies that outages are permitted for up to 12 hours for maintenance, tests, etc. This is considered as a restrictive requirement when compared to the existing Technical Specification which allows outages within bounds permitted by the action statements.

3/4.3.3.8 - Radioactive Gaseous Effluent Monitoring Instrumentation

The proposed RTS Section 3.3.3.8 has been prepared by converting existing Technical Specification Section 7/8.2.2 to a format consistent with the W STS. The proposed RTS is the same as the existing sections except as discussed below:

1. There is a footnote on the applicability statement that restricts outages up to 12 hours for maintenance, tests, etc. This is more restrictive than the existing Technical Specification which allows outages within the bounds permitted by the action statements.

2. The action statement for Item 1(b) and 1(c) of the proposed Table 3.3-10 requires that auxiliary sampling be initiated within 12 hours of a channel being declared inoperable. This requirement is not in the existing Technical Specifications.

Section 3/4.3.4 - Internal Flood Protection

The proposed RTS Section 3.3.4 has been prepared by converting the existing Technical Specification Sections 3.21 and 4.14 to a format consistent with the W STS and is same as the existing sections except as discussed below:

1. The proposed surveillance requirement (4.3.4(a)) requires a 12-hour channel check. There is no corresponding requirement in the existing Technical Specification.

Significant Hazards Consideration

In accordance with 10CFR50.92, CYAPCO has reviewed the proposed RTS sections and has concluded that they do not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed RTS do not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The determination of whether or not a proposed change is equivalent, more restrictive (or a new requirement), or less restrictive is based on the Limiting Condition for Operation and Applicability Requirements since it is these requirements which will impact the design basis accidents. In general, the conversion to the W STS yields more extensive and/or restrictive Action and Surveillance Requirements. As described above, most of the changes are more

restrictive in that they are a conservative change and there are no comparable requirements in the existing Technical Specifications. This will help ensure the operability and reliability of the systems covered under the proposed RTS. For the few changes that are less restrictive, justification is provided for the changes. Based upon the above discussion, the proposed RTS will not increase the probability or consequences of any accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously evaluated. Since there are no hardware modifications associated with the proposed changes, the performance of safety-related systems remains unaffected during operations. The operability requirements are increased over the current requirements thus enhancing the performance of safety systems. Therefore, the proposed RTS will not modify the plant response to the point where it can be considered a new accident nor are any credible failure modes created.

3. Involve a significant reduction in a margin of safety. Because the changes proposed herein provide acceptable results for the design basis accident, no additional burden will be placed on the protective boundaries for postulated accidents. In addition, there are no plant modifications associated with these changes and hence, there is no direct impact on the protective boundaries. The proposed RTS do not affect the safety limits of the protective boundaries and the bases of the proposed RTS have been modified to reflect the proposed changes.

Docket No. 50-213
B13269

Attachment 3

Technical Specification Comparison Matrix

June 1989

TECHNICAL SPECIFICATION COMPARISON MATRIX

Introduction

The Technical Specification Comparison Matrix (TSCM) was prepared to facilitate the revision of the existing Haddam Neck Technical Specifications (T.S.). The TSCM is set up denoting the proposed Technical Specification section numbers in the left hand column followed by a short description. The next column lists the corresponding existing T.S. section number. The final two columns compare the requirements contained in the proposed section with the existing T.S. and the Westinghouse STS, respectively. The key at the bottom of each page provides an explanation for the symbols located in the two comparison columns. The equivalent notation "E" may either denote that exact wording has been transposed from the existing T.S. or different verbage conveying equivalent requirements has been used. In many cases, there was not a one-for-one relationship, but rather multi-section relationships, whereas the requirements in a given T.S. section may be divided between several different sections in the proposed Technical Specification. The additional requirement notation "++" denotes that the proposed Technical Specification is more restrictive because it is an entirely new requirement as compared to the existing T.S. or it is more restrictive in the sense that the existing T.S. requirements have been changed such that they are more restrictive. This matrix is provided in a summary fashion and highlights the more significant changes. A detailed comparison in terms of additional requirements and/or less restrictive requirements is provided in Attachment 2 of this submittal.

2.0 Safety Limits and Limiting Safety Settings and 3/4.3 INSTRUMENTATION

TECHNICAL SPECIFICATION

COMPARISON MATRIX

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
2.0	Safety Limits and Limiting Safety System Settings			
2.1	Safety Limits	2.2	E	E
2.1.1	Reactor Core	2.2-1	E	E
	Applicability	2.2-1	E	E
	Action	2.2-1	E	E
2.1.2	Reactor Coolant System Pressure	2.3	E	E
	Applicability	2.3	E	E
	Actions	2.3	E	E
2.2	Limiting Safety System Setting			
2.2.1	Reactor Trip System Instrumentation Setpoints	2.4	++	E
	Applicability	2.4	++(11)	E
	Action	2.4	++	E
Table 2.2-1	Reactor Trip System Instrumentation Setpoints ++(35)			
Item 1	Manual Reactor Trip	-	++	E
Item 2	Power Range, Neutron Flux	Section 2.4 Item 4	E	E
Item 3	Intermediate Range Neutron Flux	Section 2.4 Item 8	E	E
Item 4	Pressurizer Pressure - Variable low	Section 2.4 Item 3	E	E

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
Item 5	Pressurizer Pressure - High	Section 2.4 Item 1	E	E
Item 6	Pressurizer Water Level - High	Section 2.4 Item 2	E	E
Item 7	Reactor Coolant Flow - Low	Section 2.4 Item 5	E	E
Item 8	Steam Flow - High	Section 2.4 Item 7	E	++
Item 9	Steam Generator Water level low	-	++	++
Item 10	Undervoltage - Reactor Coolant Pump	-	++	E
Item 11	Safety Injection	-	++	E
Item 12	RCP Breaker Position - Open	-	++	
Item 13	Main Steam Line Trip Valve Closure	-	++	++
Item 14	Turbine Trip	-	++	E
Item 15	Reactor Trip System Interlocks	-	++	E
Item 16	Reactor Trip System Breakers	-	++	E
3.3.1	Reactor Trip System Instrumentation			*(13)
Table 3.3-1				
Item 1	Manual Reactor Trip	Table 3.9-1 Item 8	++(1)	E
Item 2	Power Range, Neutron Flux	Table 3.9-1 Item 1	++(2), *(36)	E
Item 3	Intermediate Range, Neutron Flux	Table 3.9-1 Intermediate Range SUR Reactor Trip	++, *(36)	E
Item 4	Pressurizer Pressure - Variable Low	Table 3.9-1 Item 2	E	E

T.S.#	Description	Existing T.S.#	Comparison With Existing T.S.	Comparison With W STS
Item 5	Pressurizer Pressure - high	Table 3.9-1 Item 3	(3)	E
Item 6	Pressurizer Water level - high	Table 3.9-1 Item 4	++(1) and *(36)	E
Item 7	Reactor Coolant Flow	Table 3.9-1 Item 5	E	*(4)
Item 8	Steam Flow High	Table 3.9-1 Item 10	++	++(6)
Item 9	Steam Generator Water Level - Low Coincident with steam/feed-water flow mismatch	Table 3.9-1 Item 9	++	*(8)
Item 10	Undervoltage - Reactor Coolant Pump	-	++(7)	(9)
Item 11	Safety Injection		E	E
Item 12	Reactor Coolant Pump Breaker Position Trip	-	++(7)	++(6)
Item 13	Steam Line Isolation Valve Closure	Table 3.9-1 Item 10	E	++(6)
Item 14	Turbine Trip	-	++(7)	E
Item 15	Reactor Trip System Interlocks	-	++(7)	*(10)
Item 16	Reactor Trip System Breakers	-	++(7)	E
	Applicability	Table 3.9-1	++(11)	E
	Action	Table 3-9-1	++(12)	*(24)
4.3.1.1	Demonstrated Operable			
Table 4.3-1				
Item 1	Manual Reactor Trip	-	++	E

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
Item 2	Power Range, Neutron Flux	Table 4.2-1 Item 1	++(14)	++(15)
Item 3	Intermediate Range Neutron Flux	Table 4.2-1 Item 2	++(16)	E
Item 4	Pressurizer Pressure Variable low	Table 4.2-1 Item 8	++(17)	*(18)
Item 5	Pressurizer Pressure - High	Table 4.2-1 Item 7	E	*(18)
Item 6	Pressurizer Water Level - High	Table 4.2-1 Item 6	E	*(18)
Item 7	Reactor Coolant Flow - Low	Table 4.2-1 Item 5	E	*(19)
Item 8	Steam Flow - High		++(7)	++(6)
Item 9	Steam Generator Water Level - Flow mismatch	Table 4.2-1 Item 12	++(17)	*(19)
Item 10	Undervoltage - Reactor Coolant Pump	-	++(7)	*(19)
Item 11	Safety Injection	-	++(7)	E
Item 12	Reactor Coolant Pump Breaker Position Trip	-	++(7)	++(6)
Item 13	Main Steam Line Trip Valve Closure	-	++(7)	++(6)
Item 14	Turbine Trip	-	++(7)	*(20)
Item 15	Reactor Trip System Interlocks	-	++(7)	*(19)
Item 16	Reactor Trip System Breaker	-	++(7)	*(21)
3.3.2	Engineered Safety Features Actuation System Instrumentation			*(13)

Tab.e 3.3-2

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
Item 1	Safety Injection	Table 3.9-1 Items 6 and 11 3.11 H	++	E
Item 2	Steam Line Isolation	Table 3.9-1 Item 10	++	E
Item 3	Auxiliary Feedwater	3.8.B	*(36)	E
Item 4	Emergency Bus Undervoltage	-	++	E
Item 5	Containment Isolation	-	++	E
	Applicability	Table 3.8-1 and Table 3.9-1	++(11)	*(23)
	Action	Table 3.8-1 and Table 3.9-1	++(12)	*(25)
Table 3.3.3	Engineered Safety Features Actuation System (ESFAS) Instrumentation Trip Setpoints			
Item 1	Safety Injection	-	++	E
Item 2	Steam Line Isolation	-	++	E
Item 3	Auxiliary Feedwater	-	++	E
Item 4	Emergency Bus Undervoltage	-	++	E
Item 5	Containment Isolation	3.11.H	++	E
Table 4.3-2	ESFAS Instrumentation Surveillance Requirements			
Item 1	Safety Injection	Table 4.2-1 Item 7	E++	*(26)
Item 2	Steam Line Isolation	-	++	*(27)
Item 3	Auxiliary Feedwater	4.8.1.b, 4.8.3.C Table 4.2-1 Item 11	++	E
Item 4	Emergency Bus Undervoltage	-	++	E

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
Item 5	Containment Isolation	Table 4.2-1 Item 18	++	*(26)
3.3.3.1	Radiation Monitoring System LCO	-	++	E
	Applicability	-	++	E
	Action	-	++	E
4.3.3.1	Demonstrated Operable	Table 4.2-1 Item 19	E	E
3.3.3.2	Movable Incore Detectors - LCO	-	++	E(29)
	Applicability	-	++	E
	Action	-	++	E
4.3.3.2	Demonstrated Operable	-	++	E
3.3.3.3	Seismic Instrumentaion - Low	-	++	E(29)
	Applicability	-	++	E
	Action	-	++	E
4.3.3.3.1 and 4.3.3.3.2	Demonstrated Operable	-	++	E
3.3.3.3.4	Meteorological Instrumentation - LCO	-	++	E
	Applicability	-	++	E
	Action	-	++	E
4.3.3.4	Demonstrated Operable	-	++	E
3.3.3.5 Table 3.3-7	Accident Monitoring Instrumentation		++	*(30)
Item 1	Containment Pressure	Table 3.23-1 Item 1	E	E

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
Item 2	RCS Cold Leg Temp. - Wide Range	Table 3.23-1 Item 2	E	E
Item 3	RCS Hot Leg Temp.	Table 3.23-1 Item 3	E	E
Item 4	RCS Pressure Wide Range	Table 3-23-1 Item 4	E	E
Item 5	Pressurizer Water Level	Table 3.9-2	++	E
Item 6	Steam Generator Pressure	-	++	E
Item 7	Steam Generator Water Level - Narrow Range	-	++	
Item 8	Steam Generator Water Level - Wide Range	-	++	E
Item 9	RWST Level	-	++	E
Item 10	Boric Acid Tank Solution Level	-	++	E
Item 11	Auxiliary Feedwater Flow rate	Table 3.9-2 Item 2	++	E
Item 12	RCS Subcooling Margin Monitor	Table 3.23-1 Item 10	E	E
Item 13	PORV Block Valve Position	Table 3.9-2 Item 5	++	E
Item 14	Safety Valve and PORV (Acoustic Flow Monitor)	Table 3.9-2 Items 4 and 6	++	E
Item 15	Containment Water Level-Wide Range	Table 3.23-1 Item 5	E	E
Item 16	Containment Water Level Narrow Range	-	++	E
Item 17	Core Exit Thermocouples	Table 3.23-1 Item 6	E	E
Item 18	Main Stack - Wide Range Noble Gas Monitor	Table 3.23-1 Item 7	E	E

Item 19	Contaiment Atmosphere - High Range Radiation Monitor	Table 3.23-1 Item 8	E	E
Item 20	Reactor Vessel Water Level	Table 3.23-1 Item 9	E	E
	Applicability	refer to Note 30		
	Action	refer to Note 30		
4.3.3.5	Demonstrated Operable			
Table 4.3-6	Accident Monitoring Instrumentation Surveillance Requirements		++(31)	++(31)
Item 1	Containment Pressure	Table 3.23-2 Item 1	E	E
Item 2	RCS Cold Leg Temp - Wide Range	Table 3.23-2 Item 2	E	E
Item 3	RCS Hot Leg Temp - Wide Range	Table 3.23-2 Item 3	E	E
Item 4	RCS Pressure - Wide Range	Item 3.23-2 Item 4	E	E
Item 5	Pressurizer Water Level	-	++	E
Item 6	Steam Generator Pressure	-	++	E
Item 7	Steam Generator Water Level - Wide Range	-	++	E
Item 8	Steam Generator Water Level - Narrow Range	-	++	E
Item 9	RWST Level	Table 4.2-1 Item 16	++	E
Item 10	Boric Acid Tank Solution Level	Table 4.2-1 Item 15	E	E
Item 11	Auxiliary Feedwater Flow Rate	Table 4.2-1 Item 25	E	E

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
Item 12	RCS Subcooling Margin Monitor	Table 3.23-2 Item 10	E	E
Item 13	PORV Block Valve Position Indicator	Table 4.2-1 Item 28	++	E
Item 14	Safety Valve and PORV Acoustic Flow Monitor	Table 4.2-1 Items 27 and 29	E	E
Item 15	Containment Water Level Wide Range	Table 3.23-2 Item 5	E	E
Item 16	Containment Water Level Narrow Range	-	++	E
Item 17	Core Exit Thermocouple	Table 3.23-2 Item 6	E	E
Item 18	Main Stack-Wide Range Noble Gas Monitor	Table 3.23-2 Item 7	E	E
Item 19	Containment Atmosphere High Range Radiation Monitor	Table 3.23-2 Item 8	E	E
Item 20	Reactor Vessel Water Level	Table 3.23-2 Item 9	E	E
3.3.3.6	Fire Detection			
	Containment	3.22.E	E	E
	Cable Spreading Area	3.22.E	E	E
	1A Diesel Generator Room	3.22.E	E	E
	1B Diesel Generator Room	3.22.E	E	E
	Switchgear Room	3.22.E	E	E
	Containment Cable Vault	3.22.E	E	++
	Waste Disposal Building	3.22.E	E	++
	Aux. Feedwater Pump Room	3.22.E	E	++
	Primary Aux. Building	3.22.E	E	++
	Control Room	3.22.E	E	E
	Screen Well Building	3.22.E	E	++
	Spent Fuel Building	3.22.E	E	E
	PAB Charcoal Filter Bank	-	++	++
	Applicability	3.22	E	E
	Action	3.22.E.2	++(33)	E
4.3.3.6.1	Demonstrated Operable	4.15.E	++(34)++(32) *(36)	E
4.3.3.6.3	Demonstrated Operable	-	++	E

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
3.3.3.7	Radioactive Liquid Effluent Monitoring Instrumentation			
Table 3.3-9 Item 1	Gross Radioactivity Monitors (Auto)	Table 7.2-1 Item 1	E	E
Item 2	Gross Radioactivity Monitors (Non-Auto)	Table 7.2-1 Item 2	E	E
Item 3	Flow Rate Measurement	Table 7.2-1 Item 3	E	*(29)
	Applicability	Table 7.2-1	++	E
	Action	Table 7.2-1	E	E
4.3.3.7.1	Demonstrated Operable	8.2.1.1	E	E
3.3.3.8	Radioactive Gaseous Effluent Monitoring Instumentation			
Table 3.3-10	Main Stack			
Item 1a	Noble Gas Activity Monitor	Table 7.2-2 3.23	E	E
Item 1b	Iodine Sampler	Table 7.2-2	++	E
Item 1c	Particulate Sampler	Table 7.2-2 & 3.11.F.1	++	E
Item 1d	Stack Flow Rate Monitor	Table 7.2-2	E	E
Item 1e	Sampler Flow Rate Monitor	Table 7.2-2	E	E
	Applicability	Table 7.2-2	++	E
	Action	Table 7.2-2	E	E
4.3.3.8.1	Demonstrated Operable	Table 8.2-2	E	E
3.3.4	Internal Flood Protection			
Table 3.3-11	Liquid Level Instrumentation	Table 3.21-1	E	++

T.S.#	Description	Existing T.S. #	Comparison With Existing T.S.	Comparison With W STS
	Applicability	3.21	E	++
	Action	3.21	E	++
4.3.4	Demonstrated Operable	4.14	++	++

Notes
E = Equivalent Requirements
* = Less restrictive requirements
++= Additional Requirements

Section 3/4.3 Notes

(1) The proposed revised Technical Specification (RTS) requires 2 minimum channels operable instead of 1.

(2) The proposed RTS requires 3 minimum channels operable instead of 2.

(3) The proposed RTS requires 2 minimum channels operable instead of 1.

(4) The proposed RTS requires 1 minimum channel operable per loop instead of 2.

(5) Not used

(6) The W STS does not include this instrumentation in the reactor trip system.

(7) Although not included in the existing Technical Specifications, many of these instruments are included under current plant surveillance procedures.

(8) The proposed RTS requires 1 steam/feedwater flow mismatch instrumentation in each steam generator instead of 2.

(9) The proposed RTS requires 1 minimum channel per bus whereas the W STS requires 3 minimum out 4 total channels available.

(10) The W STS includes additional interlocks.

(11) The existing Technical Specification is applicable to full power operation only, while the proposed RTS includes additional power levels and modes.

(12) The proposed RTS has more detailed action statements and provides time contraints for subsequent action, while the existing Technical Specification does not.

(13) The W STS includes response time requirements for these instruments, whereas the proposed RTS does not. It is noted that the response time table will be provided after all the RPS modifications are complete.

(14) The proposed RTS adds the requirement to perform a channel calibration of the high, mid and low setpoints at each refueling.

(15) The W STS requires operational testing at least once per 31 days, while the proposed RTS requires testing once per 14 days.

(16) The channel calibration is added to the proposed RTS.

(17) The operational test is added to the proposed RTS.

(18) The W STS requires operational testing at least once per 31 days, while the proposed RTS requires testing once per 42 days.

(19) The W STS requires operational testing at least once per 31 days, while the proposed RTS requires testing at least once per 18 months.

(20) The W STS requires operational testing at each criticality, while the proposed RTS requires testing at least once per 18 months.

(21) The W STS requires operational testing once per 31 days, while the proposed RTS requires testing at each criticality.

(22) Not used.

(23) Auxiliary Feedwater initiation is less restrictive than the W STS.

(24) For the majority of instruments, the action statements are equivalent; however, for the Power Range Neutron Flux and all Pressurizer Pressure instrumentation, the proposed RTS is slightly less restrictive than the W STS.

(25) For the majority of instruments, the Action Statements are equivalent; however, the steam flow isolation (3 loops operating) and auxiliary feedwater initiation on trip of all main feedwater pumps are less restrictive, while the high containment pressure isolation is more restrictive than the W STS.

(26) The W STS requires channel checks and more frequent operational testing for the high containment pressure safety injection. Also, more frequent operational testing of the low pressurizer pressure on safety injection is required per the W STS.

(27) More frequent operational testing is required per the W STS.

(28) Not used.

(29) The W STS is not directly comparable due to significant differences between the Haddam Neck Plant system and the W design.

(30) The proposed RTS Table 3.3-7 is shown equivalent to the W STS, however, in many cases the number of channels and action statements are less restrictive, but the proposed RTS Table is consistent with the Generic Letter 83-37 and CYAPCO's submittals dated July 1, 1988 and March 1, 1989 and recently issued amendment #115. Applicability Modes are either consistent with or more restrictive than the W STS.

(31) The surveillance requirement included in the proposed RTS Table 4.3-6 are either consistent with or more restrictive than the W STS.

(32) The proposed RTS has required action if any detectors are inoperable, whereas the current T.S. requires action only if a minimum number of detectors are not available.

(33) Although the current T.S. requires a Channel Functional Test, the proposed RTS requires a Trip Actuating Device Operational Test, which is an improvement because the Trip Actuating Device Operational Test verifies the device along with the channel circuit.

(34) There is an additional surveillance requirement in the proposed RTS which covers non-supervised circuits.

(35) The trip setpoints given in the existing Technical Specification corresponds to the allowable values in Table 2.2-1. The proposed RTS also gives the actual trip setpoints which are not in the existing Technical Specifications.

(36) Refer to the discussion under relaxed requirement in Attachment 2.