Docket No. 50-213
B13402

Attachment 1

Haddam Neck Plant

Proposed Revised Technical Specifications

November 1989

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	≥7% and ≤8% of RTP*	≥5% and ≤10% of RTP*
2) Turbine First Stage Pressure	≥7% and ≤8% of RTP* at Turbine First Stage Pressure Equivalent	≥5% and ≤10% of RTP* at Turbine First Stage Pressure Equivalent
b. Low Flow Permissive, P-8		
1) Power Range, Neutron Flux	≤72% of RTP*	≤74% of RTP*
2) Turbine First Stage Pressure	≤72% of RTP* at Turbine First Stage Pressure Equivalent	≤74% of RTP* at Turbine First Stage Pressure Equivalent
16. Reactor Trip System Breakers	N.A.	N.A.
17. Source Range Start-up Rate Rod Stop	≤ 1.5 DPM**	≤ 2 DPM**

* RTP = RATED THERMAL POWER

**Decades per minute.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

a. AT least HOT STANDBY within the next 6 hours,

b. AT least HOT SHUTDOWN within the following 6 hours, and

c. AT least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

SURVEILLANCE REQUIREMENTS

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

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

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outage. The limitation of Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

SURVEILLANCE REQUIREMENTS (Con't.)

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

d. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal; and

e. At least once per 18 months by verifying that the flow path required by Specifications 3.1.2.2.a and 3.1.2.2.b deliver flow to the Reactor Coolant System.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump (centrifugal or metering) in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump (centrifugal or metering) OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump (centrifugal only) shall be demonstrated OPERABLE as required by Specification 4.5.1.c.3.

4.1.2.3.2 The above required charging pump (metering) shall be demonstrated OPERABLE at least once every 31 days by running the pump at its maximum speed.

4.1.2.3.3 One centrifugal charging pump shall be demonstrated inoperable by verifying that the control switch is in the trip pullout position at least once per 12 hours whenever the Overpressure Protection System of Specification 3.4.9.3a is required.

No change

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	2 2	$1^{(d)}$,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, 2, 3****	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

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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	11A

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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[(a)]	3a
2) Turbine First Stage Pressure	1	1	1	1[(a)]	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 2	1 1	2 2	1,2 3*, 4*, 5*	11 10
17. Source Range Start-up Rate Rod Stop	2	1	2	1[(d)]	4

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.

(d) THERMAL POWER below 10% 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,

****May be bypassed when the reactor is at least 1.5% Δk subcritical.

No change

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.

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, be in at least HOT STANDBY within 6 hours.

ACTION 11A:

With the number of OPERABLE channels one less than the minimum channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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.	1[d], 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, 2, 3**
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]

No change

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[(a)]
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*
17. Source Range Start-up Rate Rod Stop	S	N.A.	S/U (6)	N.A.	1[(d)]

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.

** May be bypassed when the reactor is at least 1.5% Δ k subcritical.

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	6(3/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

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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

No change

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	6 (3/Train)	1, 2, 3, 4	20
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

No change

TABLE 3.3-2 (Continued)

TABLE NOTATIONS

*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, 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 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 24 hours or reduce the THERMAL POWER to below 10% of RATED THERMAL POWER within the following 1 hour.

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 50% of tap setting voltage instantaneously.

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

* Trip actuating device operational test is defined as a test of each individual channel. A complete logic test will be performed on a refueling outage basis. On a monthly basis, an undervoltage condition will be initiated at the sensing device to verify the operability of the trip actuating device and verify that the associated alarm relays operate.

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the hot leg entry point completely around the U-bend to the top support of the cold leg; or an inspection from the point of entry (hot leg or cold leg) completely around the U-bend to the opposite end.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or sleeve* all tubes exceeding the plugging limit as defined in Specification 4.4.5.4.a.6 and plug all defective sleeves) required by Table 4.4-2.

4.4.5.5 Reports

a. Following the completion of each inservice inspection of steam generator tubes, a Special Report documenting the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 90 days following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or sleeved.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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* Tube sleeving shall be performed in accordance with the Connecticut Yankee Steam Generator Sleeving Report, Revision 1, transmitted by letter dated January 7, 1986.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Repair defective tubes and inspect additional 2S tubes in this S.G.*	C-1	None	N.A.	N.A.
			C-2	Repair defective tubes and inspect additional 4S tubes in this S.G.*	C-1	None
					C-2	Repair defective tubes.*
C-2	Repair defective tubes.*				C-3	Perform action for C-3 result of first sample.
			C-3	Perform action for C-3 result of first sample.	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., repair defective tubes and inspect 2S tubes in each other S.G.*	All other S.G.s are C-1.	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. is C-3.	Perform action for C-2 result of second sample.	N.A.	N.A.
		Notification to NRC pursuant to 50.72(b)(2) of 10 CFR Part 50.	Additional S.G. is C-3.	Inspect all tubes in each S.G. and repair defective tubes.* Notification to NRC pursuant to 50.72(b)(2) of 10 CFR Part 50.	N.A.	N.A.

$S = 3\% \times [\frac{N}{n}]$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

* Repair of defective tubes shall be limited to plugging with the exception of those which may be sleeved. Tubes with defective sleeves shall be plugged.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous Radioactivity Monitoring System, and

b. The Volume Control Tank Level Monitoring System and the Containment Main Sump Level (Narrow Range) Monitoring System.

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

ACTION:

a. With the Containment Atmosphere Gaseous Radioactivity Monitoring System Inoperable, operation may continue provided grab samples of the containment atmosphere are obtained and analyzed for gross noble gas activity at least once per 24 hours. Restore the inoperable monitor to OPERABLE status within 7 days or, prepare and submit a special report to the Commission pursuant to Specification 6.9.2 within 30 days outlining actions taken, cause of inoperability and plans for restoring the monitor to OPERABLE status.

b. With the Volume Control Tank Level Monitoring System inoperable, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the Containment Main Sump Level (Narrow Range) Monitoring System inoperable, restore the inoperable level monitoring system 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.

d. With both the Containment Main Sump Level (Narrow Range) Monitoring System and the Containment Atmosphere Gaseous Radioactivity Monitoring System inoperable, operation may continue provided grab samples of the containment atmosphere are obtained and analyzed for gross noble gas activity at least once per 4 hours. Restore the Containment Main Sump Level (Narrow Range) Monitoring System to OPERABLE STATUS within 72 hours from time of its initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

REACTOR COOLANT SYSTEM

d. If Residual Heat Removal Pump Seal leakage exceeds its allowable limit, monitor this leakage for at least 36 continuous operating hours. If the leak rate does not decrease or stabilize prior to reaching three liters per hour, the provisions of action statement b. are applicable.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous radioactivity monitor at least once per 12 hours;

b. Monitoring the containment sump inventory at least once per 12 hours;

c. Monitoring the Volume Control Tank level at least once per 4 hours;

d. Performance of a Reactor Coolant System water inventory balance at least once per 24 hours;

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours;

f. Monitoring the RHR Pit, HPSI, and charging pump cubicles for indications of leakage at seals, flanges, and valves at least once per 12 hours.

g. Performance of an operational leak rate test, for those portions of the HPSI, charging and RHR system outside containment used for recirculation, at least once per month. The test shall be conducted at a hydrostatic pressure corresponding to the operating pressure under accident conditions.

h. Monitoring leakage through each of the following ECCS valves (SI-CV-862 A, B, C, and D and SI-CV-872 A and B):

1) At least once per 18 months,

2) Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

3) Prior to returning the valve to service following maintenance, repair, or replacement work on the valve, and

4) Within 24 hours following valve actuation due to flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The total steam generator tube leakage limit of 0.4 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 0.4 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The 12-hour surveillance for leakage monitoring using the containment atmosphere gaseous radioactivity and containment sump inventory indications is a qualitative check of these indications for obvious signs of an increase in RCS leakage. The 4-hour surveillance for leakage monitoring using the VCT level indication is a quantitative determination of RCS leakage. Since constant RCS conditions are required for a valid calculation, this quantitative determination is required during periods of constant reactor power and RCS temperature only. During RCS transients, the VCT level will be monitored at least once per four hours but no quantitative determination is required.

The 12-hour surveillance requirement for leakage monitoring for RHR, HPSI, and charging systems involves general inspection of the RHR pit, HPSI, and charging pump cubicles for any obvious signs of leakage.

The limitation on the combined leakage from the RHR System, HPSI, and charging systems provides an indication of impending seal failures. The radiological consequences associated with the combined leakage is acceptable. This leakage will be considered as a portion of the IDENTIFIED LEAKAGE.

The specified allowable leakage from the listed ECCS valves is sufficiently low to ensure early detection of possible check valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for these ECCS valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from these ECCS valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

No change

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without

having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-2, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-2 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-2 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor

3/4.4.8 SPECIFIC ACTIVITY (Continued)

coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-3, 3.4-4 and 3.4-5 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-3, 3.4-4 and 3.4-5 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively.

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves", April 1975.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 22 effective full power years (EFPY) of service life. The 22 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and chemical content of the material in question, can be predicted. The heatup and cooldown limit curves of Figures 3.4-3, 3.4-4, and 3.4.5 include predicted adjustments for this shift in RT_{NDT}.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capability of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, delta RT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp (0.0145(T-RT_{NDT} + 160)) \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \quad \text{less than or equal to } K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,
K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

PRESSURE/TEMPERATURE LIMITS (Continued)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEMS

The OPERABILITY of two spring-loaded relief valves (SLRVs) or an RCS vent opening of greater than 7 square inches ensures that the RCS will be protected from presure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 315°F. Either SLRV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 20°F above the RCS cold leg temperatures, or (2) the start of a charging pump (centrifugal) and its injection into a water-solid RCS.

The Maximum Allowed SLRV Setpoint for the Low Temperature Overpressure Protection System (OPS) is derived by analysis which models the performance of the OPS assuming various mass input and heat input transients. Operation with a SLRV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the SLRV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one charging pump (centrifugal or metering) while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 20°F above RCS cold leg temperature.

The Maximum Allowed SLRV Setpoint for the OPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the RCS that could inhibit natural circulation core

3/4.4.11 REACTOR COOLANT SYSTEM VENTS (Continued)

cooling. The OPERABILITY of at least one RCS vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the RCS vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capability and testing requirements of the RCS vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period,

c. A maximum Reactor Coolant System (RCS) temperature difference of 200°F, and

d. Greater than or equal to 70°F whenever pressurizer pressure is greater than 500 psig.

APPLICABILITY: At all times.

ACTION: With the pressurizer temperature limits in excess of any of the above limits:

a. Restore the temperature to within the limits within 30 minutes,

b. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer, and

c. Determine that the pressurizer remains acceptable for continued operation.

Otherwise , be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The pressurizer temperatures for heatup and cooldown shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.9.2.2 The RCS temperature differential shall be determined to be within the limit at least once per 12 hours.

4.4.9.2.3 The pressurizer pressure shall be determined to be less than 500 psig at least once per hour whenever the pressurizer is not drained and the pressurizer temperature is less than 70°F.

SURVEILLANCE REQUIREMENTS

4.5.1 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated conditions:

Valve Number	Valve Function	Valve Position
RH-FCV-602	RHR Heat Exchanger Bypass Line	Blocked closed. Operator air supply isolated.
RH-FCV-796	RHR Heat Exchanger Discharge Line	Blocked open position. Operator air supply isolated.
RH-MOV-22	Containment Sump Suction Line	Closed. Manual Operator is locked.
SI-MOV-24	RWST Line	Locked open. Operator circuit breaker locked open.
RH-MOV-874	RHR Recirculation Line	Locked closed. Operator circuit breaker locked open.
SI-MOV-854A	HPSI Pump Suction Line	Open. Manual Operator is locked.
SI-MOV-854B	HPSI Pump Suction Line	Open. Manual Operator is locked.
SI-MOV-901	RHR/HPSI Crosstie	Closed. Manual Operator is locked.
SI-MOV-902	RHR/HPSI Crosstie	Closed. Manual Operator is locked.
SI-MOV-903	HPSI Miniflow	Open. Manual operator is locked.
SI-MOV-904	HPSI Miniflow	Open. Manual Operator is locked.

b. On start-up prior to entering Mode 4:

VALVE NO.	LOCATION	ACTION
SI-V-905	HPSI Loop 1 Injection Line	Valve blocked and locked in throttled position.
SI-V-906	HPSI loop 2 injection line	Valve blocked and locked in throttled position
SI-V-907	HPSI loop 3 injection line	Valve blocked and locked in throttled position.

TABLE 4.5-1

SAFETY INJECTION ACTUATED AUTOMATIC VALVES

VALVE NUMBER	SI POSITION
SI-MOV-861A	Open
SI-MOV-861B	Open
SI-MOV-861C	Open
SI-MOV-861D	Open
SI-MOV-871A	Open
SI-MOV-871B	Open
BA-MOV-373	Open
BA-MOV-32	Open
LD-MOV-200	Closed
LD-TV-230	Closed
CH-MOV-257	Closed
CH-MOV-257B	Closed
CH-SOV-242	Closed
CH-SOV-242B	Closed

TABLE 4.5-2

ECCS MANUAL VALVES

Valve Number
SI-MOV-24
SI-MOV-873
SI-MOV-854A
SI-MOV-854B
SI-MOV-901
SI-MOV-902
SI-MOV-903
SI-MOV-904
RH-MOV-33A
RH-MOV-33B

SURVEILLANCE REQUIREMENTS

4.5.2 a. The ECCS subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of Specification 3.5.1 with the exception that, for valves RH-FCV-602 and RH-FCV-796, restoration of valve controls be allowed.

b. One centrifugal charging pump and both High Pressure Safety Injection pumps shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the RCS cold legs of an unisolated loop is less than or equal to 315°F and the RCS is not vented by a minimum opening of 3 inches (nominal diameter) or its equivalent by verifying:

1) That the High Pressure Safety Injection pump motor circuit breakers are racked out and the cabinets locked,

2) That High Pressure Safety Injection pump discharge valves SI-V-855A and SI-V-855B are closed and locked, and

3) That the inoperable centrifugal charging pump's control switch is in the trip pullout position and red tagged, "Do Not Operate."

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to La, 0.18 % by weight of the containment air per 24 hours at Pa, 39.6 psig, or

b. A combined leakage rate of less than 0.60 La for all penetrations and valves subject to Type B and C tests, when pressurized to Pa.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 La or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 La, restore the overall integrated leakage rate to less than 0.75 La and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 La prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule, and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than Pa, 39.6 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet 0.75 La, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La at which time the above test schedule may be resumed or a corrective action plan may be prepared and submitted to the NRC that provides an acceptable alternative contingent on NRC approval.

c. The accuracy of each Type A test shall be verified using the relationship:

$$(L_{TM} + L_o - 0.25\ L_a) \leq L_c \leq (L_{TM} + L_o + 0.25\ L_a)$$

where:

L_{TM} is the percent measured containment leakage per 24 hours at pressure P_t,

L_o is the percent superimposed leakage,

L_c is the percent leakage obtained from the supplemental test result, and

L_a is replaced with L_t for reduced pressure tests.

d. Type B and C tests shall be conducted at intervals no greater than 24 months and at a pressure not less than Pa, 39.6 psig, using halogen gas detection, soap bubble, pressure decay, or other methods of equivalent sensitivity, except for tests involving:

1) Air locks, and

2) Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.9.9;

g. The provisions of Specification 4.0.2 are not applicable for Specifications 4.6.1.2.a through 4.6.1.2.d.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator of an unisolated reactor coolant loop shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided that within 4 hours the inoperable valve is restored to OPERABLE status; otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three reactor coolant loops and associated steam generators in operation and one or more main steam line Code safety valves associated with an operating loop inoperable, operation in MODES 1, 2, and 3 may proceed provided that within 4 hours the inoperable valve is restored to OPERABLE status; otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 In addition to the requirements of Specification 4.0.5, each main steam line code safety valve associated with each steam generator shall be demonstrated OPERABLE by checking its setpoint each refueling. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 The Fire Water Supply/Distribution System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by starting each pump and operating it with flow. The diesel pump shall run for 30 minutes and the electric pump for 5 minutes.

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured is in its correct position,

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

d. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and

   1) Verifying that each pump develops at least 2500 gpm at a system head of greater than or equal to 100 psig at Rated Speed,

   2) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3) Verifying that each high pressure pump starts automatically and sequentially to maintain the Fire Water Supply/ Distribution System pressure greater than or equal to 75 psig.

e. At least once per 3 years by performing a flow test of the Distribution System in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by National Fire Protection Association.

4.7.6.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1) The fuel storage tank contains at least 130 gallons of fuel, and

   2) The diesel starts from ambient conditions and operates for at least 30 minutes with flow.

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path, that is not locked, sealed or otherwise secured in position, is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

   1) By performing a system functional test which includes simulated automatic actuation of the system, and:

      a) Verifying that the system alarm check/main actuation valves in the flow path actuate to their correct positions on a simulated test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   2) By visual inspection of the spray headers to verify their integrity, and

   3) By visual inspection of each nozzle to verify that there is no blockage and that each spray area is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head system and verifying the supply piping and nozzles are unobstructed.

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.3 The following High Pressure CO_2 Systems shall be OPERABLE with the minimum number of storage bottles shown each having a net weight of not less than 90% charge weight:

a. Cable Vault - 18 bottles, and

b. Primary Auxiliary Building Charcoal Filters - 8 bottles.

APPLICABILITY:

Whenever systems, structures, components, or equipment protected by the High Pressure CO_2 Systems are required to be OPERABLE.

ACTION:

a. With one or more of the above required High Pressure CO_2 Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for the unprotected equipment and/or area.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.3 Each of the above required High Pressure CO_2 Systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying CO_2 storage bottle weight to be at least 90% of full charge weight, and

b. At least once per 18 months by:

1) Verifying that all system components operate properly when manual devices are manually operated and automatic devices are actuated by a simulated signal, and

2) Performance of an air flow test through system headers and nozzles to assure no blockage.

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LIMITING CONDITIONS FOR OPERATION

3.7.6.4 The following Halon 1301 Systems shall be OPERABLE with the minimum number of storage containers shown each having a net weight of not less than 95% and a pressure of not less than 324 psig:

a. Switchgear Room - 8 containers,

b. Control Room - 3 containers.

APPLICABILITY: Whenever systems, structures, components, or equipment protected by the Halon System are required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon 1301 Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for the unprotected equipment and/or area.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.4 Each of the above required Halon 1301 Systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying each Halon storage container net weight to be at least 95% of full charge weight and with a pressure of not less than 324 psig (adjusted for temperature), and

b. At least once per 18 months by:

1) Verifying that all system components operate properly when manual devices are manually operated and automatic devices are actuated by a simulated signal, and

2) Performance of an air flow test through headers and nozzles to assure no blockage.

3/4.7.7 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.7 All fire rated assemblies (walls, floor/ceiling, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems required OPERABLE for safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times unless otherwise determined that the separation of safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area is not required based on the MODE of operation.

ACTION:

a. With one or more of the above required fire rated assemblies and/or penetration sealing devices inoperable, within 1 hour:

   1. Determine that the fire areas/zones on both sides of the affected fire rated assembly and/or penetration sealing device are monitored by either an OPERABLE fire detection or automatic suppression system at the fire barrier and establish a fire watch patrol that inspects both areas at least once per hour, or

   2. Establish a continuous fire watch on at least one side of the affected fire rated assembly and/or penetration seal.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.1 At least once per 18 months, the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly, In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met.

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.7 FIRE RATED ASSEMBLIES

The functional integrity of the fire rated assemblies and barrier penetrations ensures that fires will be confined (for a time of at least equal to the minimum time rating of the associated fire barrier) from spreading to adjacent portions of the facility. These design features minimize the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishing of the fire. The fire barrier penetrations are a passive element in the facility Fire Protection Program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition.

During periods of time when a barrier is not functional, alternate measures are taken to prevent the possible spread of fire. These measures include verifying the operability of fire detection or suppression systems on <u>both</u> sides of the affected barrier and establishing a fire watch patrol, or posting a continuous fire watch in the vicinity of the affected barrier.

3/4.7.8 FLAMMABLE LIQUIDS CONTROL

The control of flammable liquids in the control room substantially reduces any fire loadings in the control room. Specification 3/4.7.8 also satisfies a NRC condition of an Appendix R control room exemption issued November 14, 1984.

No change

PLANT SYSTEMS

3/4.7.11 PRIMARY AUXILIARY BUILDING AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11 The Primary Auxiliary Building Air Cleanup System shall be OPERABLE and in operation.

APPLICABILITY: MODES 1, 2, 3, and 4. Also, during operations involving movement of fuel assemblies or control rods within the containment.

ACTION:

a. With the Primary Auxiliary Building Air Cleanup System inoperable during MODES 1, 2, 3 and 4 restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the Primary Auxiliary Building Air Cleanup System inoperable or not in operation during operations involving movement of fuel assemblies or control rods within the containment, suspend all operations involving movement of fuel assemblies or control rods within the containment.

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

SURVEILLANCE REQUIREMENTS

4.7.11 The Primary Auxiliary Building Air Cleanup System shall be demonstrated OPERABLE and in operation:

a. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following major painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is greater than 20,000 cfm;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10% at test conditions of 86°F, 95% relative humidity, atmospheric pressure, and 40 feet/min. face velocity in accordance with ASTM D3803; and

3) Verifying a system flow rate of greater than 20,000 cfm during system operation when tested in accordance with ANSI N510-1980.

4) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 1.6 inches Water Gauge while operating the system at a flow rate of 20,000 cfm ±10%.

b. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of greater than 20,000 cfm; and

c. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of greater than 20,000 cfm.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 10% at test conditions of 86°F, 95% relative humidity, atmospheric pressure and 40 feet/min. face velocity in accordance with ASTM D3803.

e. At least once per 18 months, verify a system flowrate of greater than 20,000 cfm during system operation aligned for Containment Purge.

3/4.7.9 FEEDWATER ISOLATION VALVES

The accident analysis for a main steam line break assumes that the main feedwater isolation valves will close on a containment isolation actuation signal (CIAS). Also, the closure of these valves based on a CIAS is credited in determining the Pressure/Temperature limits for the purpose of environmental qualification. The feedwater isolation valves act as a backup to the feedwater regulation valves in the event a feedwater regulation valve fails open during a Main Steam Line Break.

3/4.7.10 EXTERNAL FLOOD PROTECTION

The thresholds regarding flood protection ensure that facility protective actions will be taken (and the orderly shutdown of the plant to MODE 3 will be made) in the event of flood conditions. The estimated Connecticut River probable maximum flood (PMF) level, including wave effects (i.e., still water level), is 39.5 feet mean sea level. Normal flood control measures provide protection to safety-related equipment to El. 30 feet mean sea level. Normal flood protection to this elevation is based on a low probability of exceedance and structural capacity limitations. Based on the one to two day rise period of the PMF, alternative means of providing decay heat removal for flooding events up to the PMF is provided in AOP 3.2-24.

3/4.7.11 PRIMARY AUXILIARY BUILDING (PAB) AIR CLEANUP SYSTEM

PAB Air Cleanup System consists of two exhaust fans, two prefilters, a HEPA-HECA filter assembly, and interconnecting ductwork.

Air cleanup is accomplished using one exhaust fan, one prefilter, the HEPA-HECA filter, and interconnected ductwork.

The radiological consequences analyses for loss-of-coolant accidents assume Primary Auxiliary Building efficiencies which are ensured by this Technical Specification. Also, in consideration of a fuel handling accident inside containment, (i.e., when the containment is being purged) the purge discharge would be directed through the Primary Auxiliary Building charcoal filters. Credit is again taken for these filters in reducing the radiological consequences.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch installed and secured in place,

b. A minimum of one door in the airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be, either:

   1) Closed by an isolation valve, blind flange, manual valve, or special device, or

   2) Be capable of being closed by an OPERABLE containment purge supply, purge exhaust, or purge exhaust bypass isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building, other than placing an irradiated fuel assembly into a safe storage location.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE containment purge supply, purge exhaust, or purge exhaust bypass isolation valve or capable of being closed by an OPERABLE airlock door within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations referred to in Specification 3.9.4c are in their closed/isolated condition,

b. Testing the containment purge supply, purge exhaust, and purge exhaust bypass isolation valves in accordance with Specification 4.9.9, and

c. Verifying that at least one of the doors in the airlock is capable of being closed.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1650 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Administrative controls that prevent the travel of loads in excess of 1650 pounds over fuel assemblies shall be in place prior to lifting a load in excess of 1650 pounds.

3/4.9.9 CONTAINMENT PURGE SUPPLY, PURGE EXHAUST, AND PURGE EXHAUST BYPASS ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge supply, purge exhaust, and purge exhaust bypass isolation valves shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building, other than placing an irradiated fuel assembly into a safe storage location.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge supply, purge exhaust, and purge exhaust bypass isolation valves shall be verified OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS and prior to movement of irradiated fuel within containment by verifying that containment purge, bypass or exhaust isolation valves are accessible for manual operation or verifying that the associated penetrations are blind flanged.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 and 3/4.5.2 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements. In Mode 4, the RHR heat exchanger(s) may be considered OPERABLE while aligned to the Component Cooling System.

The limitation for a maximum of one centrifugal charging pump and one metering pump to be OPERABLE and the Surveillance Requirement to verify the remaining centrifugal charging pump and high pressure safety injection pumps to be inoperable below 315°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single low temperature overpressurization relief valve.

In order to use the HPSI pumps to provide high pressure recirculation following a small break loss of coolant accident (LOCA) coincident with a single active failure, the following modifications to the emergency core cooling system have been made. A piping crosstie between each HPSI pump suction and the RHR pump discharge has been installed. Two valves, SI-MOV-901 and SI-MOV-902 have been installed in this crosstie. The two manual HPSI pump suction valves have been replaced with motor-operated valves, SI-MOV-854A and B, to prevent contaminated water from entering the RWST when using the HPSI pumps to provide flow to the core during recirculation. Two MOVs (SI-MOV-903 and 904) in series have been added to the HPSI miniflow line to provide for redundant isolation of the RWST during sump recirculation.

The manual core deluge isolation valve has been replaced with a de-energized motor-operated valve, SI-MOV-873. This valve is locked open with the breaker locked open to ensure that adequate flow is available to the core deluge system. The valve may be energized and closed if a core deluge valve failed to close during transfer to sump recirculation.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that as a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

Surveillance requirements for throttle valve position and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA.

No change

AUXILIARY FEEDWATER SUPPLY (Continued)

In addition, the auxiliary feedwater system can be initiated manually. In this case, feedwater is available from the DWST by gravity feed to the auxiliary feedwater pump. The specified 50,000 gallons of water in the DWST is adequate for decay heat removal for a period of at least 2 hours. Within this period, decay heat removal demands are reduced to approximately 150 gpm. Makeup water is available during this period from the PWST which contains a minimum volume of 80,000 gallons. The PWST transfer pumps can transfer 200 gpm from the PWST to the DWST. An alternate supply can be provided from the 100,000 gallons Recycled Primary Water Storage Tank.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 0.4 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE TRIP VALVES

The OPERABILITY of the main steam line trip valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam line trip valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RTNDT of 10°F and are sufficient to prevent brittle fracture. The heatup and cooldown rate of 100°F/hr for the steam generators are specified to ensure that stresses in these vessels are maintained within acceptable design limits.

3/4.7.3 SERVICE WATER SYSTEM

The OPERABILITY of the Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analysis. A service water header is comprised of the two service water pumps associated with each diesel generator and the safety-related piping and components. The service water headers may be tied together by an open service water header cross-connect in the intake structure.

3/4.7.4 SNUBBERS

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

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18-month intervals. These tests will include stroking of snubbers to verify freedom of movement over the full stroke, restraining characteristics, and drag force (if applicable). Ten percent (10%) of the total of each type of snubber represents an adequate sampling for these tests. Observed failures on these samples require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance program.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1. The minimum distance to the boundary of the exclusion area as defined in 10CFR100.3 shall be 1740 feet.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a hemispherical dome roof and a flat base, and having the following design features:

a. Nominal inside diameter of cylinder = 134 feet - 11.25 inches.

b. Nominal inside height of cylinder = 119.5 feet (Not including dome).

c. Nominal inside height of containment = 188 feet - 11.5 inches (From containment floor to inside top of dome).

d. Minimum thickness of concrete walls = 4.5 feet.

e. Minimum thickness of concrete dome = 2.5 feet.

f. Minimum thickness of concrete bottom mat = 9 feet.

g. Nominal thickness of steel liner = 1/4 to 1/2 inch (bottom = 1/4 inch, spherical dome = 1/2 inch and side wall = 3/8 inch).

h. Net free volume = 2.232×10^6 cubic feet (nominal).

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 40 psig and a temperature of 260°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rod locations. Fuel rod locations may at any time during plant life consist of one of the following: (1) fuel rods clad with Type 304 stainless steel, (2) filler rods fabricated from Type 304 stainless steel, or (3) vacancies as evaluated by the cycle-specific reload analysis. Each fuel assembly shall have a nominal active fuel length of 120.5 inches and contain a typical weight of 411.5 kilograms uranium. The core loading shall have a maximum enrichment of 4.00 weight percent U- 235.

Docket No. 50-213
B13402

Attachment 2

Haddam Neck Plant

Description of the Proposed Technical Specification Changes
and Significant Hazards Consideration Discussion

November 1989

Haddam Neck Plant
Description of The Proposed Technical Specification Changes and Significant Hazards Consideration Discussion

Description of the Proposed Changes

References (1) through (11) provided a complete set of the proposed revised Technical Specifications (RTS) for the Haddam Neck Plant. All sections of the existing Technical Specifications for the Haddam Neck Plant have been reformatted and upgraded except for Sections 3.6, 3.12, and 4.5 [References (1) through (6)]. These sections, 3.6, 3.12, and 4.5, have been reformatted and upgraded and are included in References (8) and (11). In addition, the changes included in References (8) and (11) reflect the modification completed in the 1989 refueling outage. The Technical Specification changes included in Reference (7) reflect the upgrade of the reactor protection system and the nuclear instrumentation. The Technical Specification changes included in Reference (9) reflect Cycle 16 reload changes and, lastly, the Technical Specification changes included in Reference (10) reflect the changes associated with the fire detection/suppression system upgrades in support of the new switchgear building. The following changes are proposed to the RTS primarily to incorporate the NRC's comments to the proposed RTS.

1. Tables 2.2-1, 3.3-1, and 4.3-1, Item 17, Source Range Start-Up Rate Rod Stop

   The existing Technical Specifications (TS) Table 3.9-1 requirement for the source range start-up rate rod stop is reinstated in the proposed RTS. Corresponding changes are incorporated in Table 3.3-1, Reactor Trip System Instrumentation and Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements. The changes will make the proposed RTS consistent with the existing TS. However, the changes proposed in Reference (7) will supersede this change. The proposed changes provided herein are only for record purposes. No design basis accident analysis takes credit for the source range start-up rate rod stop. As such, no design basis accidents are adversely affected due to the changes.

2. Section 4.0.2 and Corresponding Bases Section

   The proposed change to the requirements of Specifications 4.0.2 will remove any unnecessary restriction on extending the surveillance requirement and will result in a benefit to safety when plant conditions are not conducive to the safe conduct of the surveillance requirement. The removal of the 3.25 limit will provide greater flexibility in the use of the provision for extending surveillance intervals, reduce the administrative burden associated with its use, and have a positive effect on safety. This change is consistent with Generic Letter 89-14. No design basis accidents are adversely impacted due to the change.

3. Section 4.1.2.2, Page 3/4 1-12

The change corrects a typographical error and requires verification of flow paths from the boric acid tank via the boric acid pumps and centrifugal charging pumps to the reactor coolant system by referencing Specification 3.1.2.2.b. Thus, the surveillance requirement of Section 4.1.2.2 will become restrictive by the change. No design basis accidents are adversely affected due to the change.

4. Section 3.1.2.3, page 3/4 1-13

The change corrects a typographical error. No design basis accidents are affected due to the change.

5. Tables 3.3-1 and 4.3-1, Item 3, Intermediate Range, Neutron, Flux, High Start-Up Rate

The proposed change will also require the trip to be operable in Mode 1 below 10 percent power and increase the number of minimum channels operable from 1 to 2 in Table 3.3-1, Reactor Trip System Instrumentation. Corresponding changes to Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirement, are also proposed. As such, the proposed change provides more stringent requirements compared with the proposed RTS and is equivalent to the existing TS. However, the changes proposed in Reference (7) will supersede this change. The proposed change provided herein is only for record purposes. No design basis accident is adversely affected due to the change.

6. Tables 3.3-1 and 4.3-1, Item 6, Pressurizer Water Level-High

The change will require the trip to be operable in Modes 1, 2, and 3 instead of Mode 1 only. In addition, the change will allow the high pressurizer event trip to be bypassed when the reactor is at least 1.5 percent ΔK subcritical. It is noted that the original safety evaluations performed are still applicable since the change will be consistent with the existing TS. No design basis accident is adversely affected due to the change.

7. Table 3.3-1, Item 15a, Low Power Block, P-7

The proposed change will require the trip to be operable at any power level above 10 percent of rated thermal power. The proposed RTS in the previous submittal [Reference (4)] require the trip to be operable for power above 10 percent but below 74 percent of rated thermal power. The change is consistent with the existing TS. No design basis accident is adversely affected due to the change.

8. Table 3.3-1, Item 11, Safety Injection

The proposed change adds a new action statement to be required for the safety injection system. The new action statement (Action 11A) reads as follows: "with the number of OPERABLE channels one less than the minimum channels Operable requirement, be in at least HOT STANDBY within 6 hours. However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE." This added restriction eliminates the 48-hour period for restoring the channel prior to applying the 6-hour hot standby requirement.

The change will make the proposed RTS consistent with the existing TS and with the Westinghouse STS. The change will not impact the consequences of any design basis accidents.

9. Table 3.3-1, Action 11

The change corrects a typographical error. No design basis accident is affected by the change.

10. Table 3.3-2, Items 1.b and 5.a, Containment Pressure - High; Item 3.a, Wide Range SG Water Level - Low; Item 3.b, Trip of All Main Feedwater Pumps

The proposed change will increase the number of minimum channels operable from 4 to 6 for Containment pressure-high in Table 3.3-2, Engineered Safety Features Actuation System Instrumentation. The change will also change the Action statement for Item 5.a to be consistent with the existing TS. For Items 3.a and 3.b, the change requires the allowable restoration time for an inoperable channel to be reduced from 48 hours to 24 hours. The 24-hour time is based on engineering judgment to take actions to repair the inoperable channel. The change provides more stringent requirements relative to system's surveillance and operation. No design basis accidents are adversely impacted due to the change. It is noted that the proposed change to Table 3.3-2, Items 1.b and 5.a, will be superseded by the changes proposed in Reference (7).

11. Table 3.3-3, Footnote to Item 4.a, 4.16 kV Bus Undervoltage - Level 1; Table 4.3-2, Footnote to Items 4.b and 4.c, 4.16 kV Bus Undervoltage - Level 2 and Level 3

Table 3.3-3, Engineered Safety Features Actuation System Instrumentation Trip Setpoints of the proposed RTS in the previous submittal [Reference 8)] contains the description of test parameters for the 4.16-kV Bus Level 1 undervoltage setpoint and minimum allowable value. This description states that the device must change state within .95 - 1.05 seconds when the input voltage to the device goes from normal to zero volts instantaneously.

The proposed change requires that the relays actuate when the input voltage decreases instantaneously from normal to 50 percent of the tap setting voltage. The 4.16-kV Bus Level 1 undervoltage allowable value is the minimum voltage to which the safety bus may degrade before all required loads must be transferred from offsite to onsite power sources. The table notation for the allowable value further defines the operation of the undervoltage relays and its actuation parameters. By requiring the device to change state within one second ± 5 percent when the input voltage to the device reduces from normal to 50 percent of tap setting voltage instantaneously, the relay is being challenged to operate in a real degraded voltage situation. Previously, allowing the input voltage to the device to drop to zero did not fully test the time-voltage characteristics of the induction coil in the degraded voltage range because loss of all voltage just causes the relay to return to its de-energized state. The change represents a more conservative test and is consistent with the plant's standard method of testing undervoltage relays of this type.

The proposed change also adds the footnote (*) to the frequency at which the 4.16-kV bus undervoltage Level 2 and Level 3 trip actuating device operational test is specified. Adding the footnote defines the monthly surveillance as a test of each individual channel rather than a test of the logic. the Level 1 trip actuating device operational test currently has this footnote.

In order to test the complete logic associated with these undervoltage trip relays, two channels would need to be placed out of service. This would be in direct conflict with Action Statement 24 of Table 3.3-2, which allows only one channel to be bypassed for up to 2 hours for surveillance testing with one channel inoperable. It is noted that a complete undervoltage relay calibration and logic check with still be performed during refueling and is not affected due to the change.

Based on the above, the changes do not impact the consequences of any design basis accidents and the probability of failure of a safety system associated with these changes is not increased.

12. <u>Section 4.4.5.5, Reports, and Table 4.4.2, Steam Generator Tube Inspection</u>

The proposed change adds words "or sleeved" to Section 4.4.5.5.a and deletes a note to Table 4.4-2 to make the proposed RTS consistent with the existing TS. This eliminates the ability to take exception to a requirement and is thus more restrictive. The change does not impact the consequence of any design basis accidents.

13. Section 3.4.6.1.b and Action a

The change corrects a typographical error. No design basis accident is affected due to the change.

14. Section 3.4.6.2, Operational Leakage

The proposed change revised the Surveillance 4.4.6.2.1.c (VCT level monitoring) frequency from "at least once per 12 hours" to "at least once per 4 hours." In addition, the bases section is changed to clarify "monitoring" term. As such, the proposed change provides a more restrictive surveillance requirement. No design basis accidents are adversely affected due to the change.

15. Section 3.4.9.2, Pressurizer

The proposed change deletes the surveillance requirements for the auxiliary spray (Section 4.4.9.2.2) as there is no limiting condition for operation for this surveillance. This was inadvertently included in the previous submittal [Reference (3)].

16. Sections 4.5.1 and 4.5.2, ECCS, Table 4.5-1 and Table 4.5-2

The change corrects typographical errors. No design basis accidents are affected due to the change.

17. Section 3.6.1.2, Footnote on Page 3/4 6-2, and Section 4.6.1.2.g

The proposed change deletes the footnote regarding the exemption for certain containment penetrations since CYAPCO has not received a permanent exemption to Appendix J, paragraph III.C.2.(a) for these penetrations. The change provides a more restrictive requirement. No design basis accidents are adversely affected due to the change. Section 4.6.1.2.g is revised to clarify that the exception to Specification 4.0.2 is only applicable to Sections 4.6.1.2.a through 4.6.1.2.d. No design basis accidents are affected due to this change.

18. Section 4.7.1.1, Turbine Cycle Safety Valves Surveillance

The proposed change will allow the plant to enter in Mode 3 to perform SG code safety testing following an outage. The existing TS requires the safety valves to be operable when critical. This is consistent with the plant's current practices and the existing TS. No design basis accidents are adversely affected due to the change.

19. Section 4.7.6.1.1, Item e, Fire Water Supply/Distributions Surveillance Requirement

The proposed change specifies the reference document that provides a criterion to perform a system flow test. The reference document is Chapter 5, Section 11, of the Fire Protection Handbook, 14th Edition, published by National Fire Protection Association. No design basis accidents are affected due to the change.

20. Section 4.7.6.2, Item d, Spray and/or Sprinkler Systems

The proposed change removes the option to use water to perform a flow test through each open head system. The change is consistent with the current plant practices. No design basis accidents are adversely affected due to the change.

21. Section 3.7.6.3, Action Item (a), CO_2 System

The proposed change clarifies the location and the equipment the ACTION affects. No design basis accidents are affected due to the change.

22. Section 4.7.6.3, Item b.1, CO_2 System Surveillance Requirements

The proposed change revises the wording of the surveillance to clarify that all system components will be verified. The change is editorial in nature. No design basis accidents are affected due to the change.

23. Section 3.7.6.4, Action Item (a), Halon System

The proposed change clarifies the location and the equipment the ACTION affects. No design basis accidents are affected due to the change.

24. Section 4.7.6.4, Item b.1, Halon System Surveillance Requirements

The proposed change revises the wording of the surveillance to clarify that all system components will be verified. The change is editorial in nature. No design basis accidents are affected due to the change.

25. Section 3.7.7, Action 3, Fire Rated Assemblies, and Bases Section 3/4.7.7

The proposed change deletes the option to temporarily repair inoperable fire-rated assemblies. The change is consistent with the existing TS. This change will also revise the bases section by deleting the reference to the installation of a temporary fire stop. No design basis accidents are adversely affected due to the change.

26. Section 3/4.7.11, Primary Auxiliary Building Air Cleanup System Surveillance and Bases

The change will achieve consistency among the proposed RTS by changing the documentation referenced from ANSI N510-1975 to ANSI N510-1980. The change will also revise the bases section to indicate number of fans and/or filtration system required to be operable to meet LCO of Section 3.7.11. The change is administrative in nature. No design basis accidents are affected by the change.

27. Section 3.9.4, Containment Building Penetration, LCO (b)

The proposed change will require one door in the airlock CLOSED rather than OPERABLE. This change is consistent with the current plant practices and is consistent with the Westinghouse STS. The other change proposed herein recognizes the current name of the system (i.e., Containment Purge Supply, Purge Exhaust, and Purge Exhaust Bypass System). No design basis accidents are adversely affected due to the change.

28. Section 3/4.9.7, Crane Travel-Spent Fuel Storage Pool Building

The proposed change will clarify the movement of load in excess of 1650 pounds over spent fuel assemblies in the storage pool. The change is an administrative change. No design basis accidents are affected due to the change.

29. Section 3/4.9.9, Containment Purge Supply, Purge Exhaust, and Purge Exhaust Bypass System

The proposed change recognizes the current name of the system. No design basis accidents are affected due to the change.

30. Bases Sections 3/4.5.1 and 3/4.5.2, ECCS Subsystems; 3/4.7.3, Service Water System

The proposed change adds a statement to clarify the operability of RHR heat exchangers in Mode 4 (Section 3/4.5.1 and 3/4.5.2). In addition, the change adds a statement to Bases Section 3/4.7.3, Service Water System, to indicate that the service water headers may be tied together by an open service water header cross-connect in the intake structure. As such, the change in the bases section will not affect any design basis accidents.

31. Section 5.1.1, Exclusion Area

The change adds an explicit statement to indicate the minimum distance to the boundary of the exclusion area. The change will make the proposed TS consistent with the existing TS. No design basis accidents are affected due to the change.

32. Section 5.3.1, Fuel Assemblies

Inspection of fuel assemblies during the current refueling outage has revealed pins with the fuel cladding failures. This problem is being addressed by fuel reconstitution which in many cases results in failed fuel pins being replaced with "filler rods" fabricated from Type 304 stainless steel or even left as vacancies. This fuel reconstitution will be conducted as demonstrated to be acceptable by a cycle-specific reload analysis. The change is proposed to reflect the options of having "filler rods" or "vacancies" as well as actual fuel rods. Since changes due to reconstitution will be fully evaluated prior to implementation, the extra flexibility allowed by the change to the design features section has no impact on any design basis accidents.

In general, most of the above changes are either administrative or will make the proposed RTS consistent with the existing TS. Essentially, the changes provide more restrictive requirements than the proposed RTS in the previous submittals.

Significant Hazards Consideration

It should be noted that the significant hazards consideration (SHC) provided below supplements the SHC discussion previously included in References (1) through (11).

In accordance with 10CFR50.92, CYAPCO has reviewed the changes to the base document 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 of occurrence or consequences of an accident previously analyzed. The proposed changes provide some additional requirements from the proposed RTS (previous submittals) to bring the specifications more consistent with the existing TS requirements. Also, the proposed changes correct typographical errors and clarify some ambiguous wording and reference documentation. These changes do not have any impact on the design basis accidents. Section 5.3.1 for fuel assemblies will now allow fuel rod locations to be filled with alternate rods as used in reconstitution. Since the changes due to reconstitution will be fully evaluated prior to implementation, the additional flexibility allowed by the changes to the Design Features Section of the RTS has no impact on the design basis accidents.

2. Create the possibility of a new or different kind of accident from any previously analyzed. Since there are no changes in the way the plant is operated, the potential for an unanalyzed accident is not created. No new failure modes are introduced.

3. Involve a significant reduction in a margin of safety. The changes do not have any adverse impact on the protective boundaries. Since the changes also do not affect the consequences of any accident previously analyzed, there is no reduction in a margin of safety.