TABLE 2.2-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 4 psig	≤ 4 psig
6. Steam Generator Pressure - Low (2)	≥ ~~570~~ 685 psia	≥ ~~570~~ 685 psia *
7. Steam Generator Water Level - Low	≥ 10 inches below top of feed ring.	≥ 10 inches below top of feed ring.
8. Axial flux offset (3)	Trip setpoint adjusted to not exceed the limit lines of Figure 2.2-1.	Trip setpoint adjusted to not exceed the limit lines of Figure 2.2-1.
9. Thermal Margin/Low Pressure (1)		
a. Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-2 and 2.2-3.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-2 and 2.2-3.
b. Steam Generator Pressure Difference - High (1)	≤ 135 psid	≤ 135 psid
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	≥ 1100 psig	≥ 1100 psig
11. Rate of Change of Power - High (4)	≤ 2.6 decades per minute	≤ 2.6 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 10^{-4}% of RATED THERMAL POWER.

TABLE 2.2-1 (Cont'd)

TABLE NOTATIONS (Cont'd)

(2) Trip may be manually bypassed below ~~685~~ 785 psia; bypass shall be automatically removed at or above ~~685~~ 785 psia. *

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 12% of RATED THERMAL POWER.

LIMITING SAFETY SYSTEM SETTINGS

BASES

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.195 1.23 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.195 1.23 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of ~~570~~ 685* psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± ~~[illegible]~~ 85* psi in the accident analyses. which was based on the Main Steam Line Break event.

BASES

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. * ~~The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 13 minutes before auxiliary feedwater is required.~~

SEE * INSERT NEXT PAGE

Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than 1.195 1.23 nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.195 1.23.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1750 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The specified setpoint in combination with the auxiliary feedwater actuation system ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event. *

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 10^{-4} of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$ of RATED THERMAL POWER.

(b) Trip may be manually bypassed below ~~605~~ 785 psia; bypass shall be automatically removed at or above ~~605~~ 785 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 12% of RATED THERMAL POWER.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by 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 and/or open the protective system trip breakers.

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 either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

ATTACHMENT 2

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8.	CVCS ISOLATION					
	a. Manual (CVCS Isolation Valve Control Switches)	1/Valve	1/Valve	1/Valve	1, 2, 3, 4	6
	b. West Penetration Room/Letdown Heat Exchanger Room Pressure - High	4	2	3	1, 2, 3, 4	7*
9.	AUXILIARY FEEDWATER Actuation System					
	a. Manual (Trip Buttons)	2 sets of 2 per S/G	1 set of 2 per S/G	2 sets of 2 per S/G	1, 2, 3	6
	b. Steam Generator Level - Low	4/SG	2/SG	3/SG	1,2,3	7
	c. Steam Generator ΔP High	4/SG	2/SG	3/SG	1,2,3	7

*

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < ~~1700~~ 1800 psia; bypass shall be automatically removed when pressurizer pressure is ≥ ~~1700~~ 1800 psia.

(c) Trip function may be bypassed in this MODE below ~~600~~ 785 psia; bypass shall be automatically removed at or above ~~600~~ 785 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, 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.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
c. Pressurizer Pressure - Low	$\geq$ ~~1570~~ 1725 psia	$\geq$ ~~1578~~ 1725 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
3. CONTAINMENT ISOLATION (CIS) #		
a. Manual CIS (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
4. MAIN STEAM LINE ISOLATION		
a. Manual (MSIV Hand Switches and Feed Head Isolation Hand Switches)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ ~~570~~ 685 psia	$\geq$ ~~[illegible]~~ 685 psia *

\# Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

ATTACHMENT 3

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES	
8. CVCS ISOLATION			
West Penetration Room/Letdown Heat Exchanger Room Pressure - High	$\leq$ 0.5 psig	$\leq$ 0.5 psig	
9. AUXILIARY FEEDWATER ACTUATION SYSTEM			
a. Manual (trip buttons)	Not Applicable	Not Applicable	
b. Steam Generator (A or B) Level - Low	-194" to -149" (inclusive)	-194" to -149" (inclusive)	
c. Steam Generator Δ P-High (SG-A > SG-B)	$\leq$ 130.0 psid	$\leq$ 130.0 psid	*
d. Steam Generator Δ P-High (SG-B > SG-A)	$\leq$ 130.0 psid	$\leq$ 130.0 psid	

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIS	
Containment Isolation	Not Applicable
d. RAS	
Containment Sump Recirculation	Not Applicable
e. AFAS	
AUXILIARY FEEDWATER ACTUATION	Not Applicable *
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 30*/30**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 30*/30**
b. Containment Isolation	≤ 30
c. Containment Fan Coolers	≤ 35*/10**
4. Containment Pressure--High	
a. Containment Spray	≤ 60*/60**
5. Containment Radiation-High	
a. Containment Purge Valves Isolation	≤ 5

ATTACHMENT 4

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	$\leq$ 6.9
b. Feedwater Isolation	$\leq$ 80
7. Refueling Water Tank-Low	
a. Containment Sump Recirculation	$\leq$ 80
8. Reactor Trip	
a. Feedwater Flow Reduction to 5%	$\leq$ 20
9. Loss of Power	
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	$\leq$ 2.2***
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	$\leq$ 8.4***
10. Steam Generator Level - Low	
a. Steam Driven AFW Pump	$\leq$ 54.5
b. Motor Driven AFW Pump	$\leq$ 54.5* / 14.5**
11. Steam Generator ΔP-High	
a. Auxiliary Feedwater Isolation	$\leq$ 20.0

*

TABLE NOTATION

* Diesel generator starting and sequence loading delays included.

** Diesel generator starting and sequence loading delays not included. Offsite power available.

*** Response time measured from the incidence of the undervoltage condition to the diesel generator start signal.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. CONTAINMENT SUMP RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Tank - Low	N.A.	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
6. CONTAINMENT PURGE VALVES ISOLATION##				
a. Manual (Purge Valve Control Switches	N.A.	N.A.	R	N.A.
b. Containment Radiation - High Area Monitor	S	R	M	6
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage	N.A.	R	M	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage	N.A.	R	M	1, 2, 3
8. CVCS ISOLATION West Penetration Room/ Letdown Heat Exchange Room Pressure - High	N.A.	R	M	1, 2, 3, 4
9. AUXILIARY FEEDWATER				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Steam Generator Level-Low	S	R	M	1, 2, 3
c. Steam Generator ΔP-High	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3

*

Containment purge valve isolation is also initiated by SIAS (functional units 1.a, 1.b and 1.c).

ATTACHMENT 6

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

	INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE	
1.	Reactor Trip Breaker Indication	Cable Spreading Room	OPEN-CLOSE	1/trip breaker	*
2.	Reactor Coolant Cold Leg Temperature	2C43	212-705^{0}F	1	
3.	Pressurizer Pressure	2C43	0-4000 psia	1	
4.	Pressurizer Level	2C43	0-360 inches	1	
5.	Steam Generator Pressure	2C43	0-1200 psig	1/steam generator	
6.	Steam Generator Level	2C43	-401 to +63.5 inches	1/steam generator	*

ATTACHMENT 7

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CALIBRATION	
1.	Reactor Trip Breaker Indication	M	N.A.	*
2.	Reactor Coolant Cold Leg Temperature	M	R	
3.	Pressurizer Pressure	M	R	
4.	Pressurizer Level	M	R	
5.	Steam Generator Pressure	M	R	
6.	Steam Generator Level	M	R	

ATTACHMENT 8

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	
1.	Containment Pressure	2	*
2.	Wide Range Logarithmic Neutron Flux Monitor	2	
3.	Reactor Coolant Outlet Temperature	2	*
4.	Pressurizer Pressure	2	
5.	Pressurizer Level	2	
6.	Steam Generator Pressure	2/steam generator	
7.	Steam Generator Level (Wide Range)	2/steam generator	*
8.	Auxiliary Feedwater Flow Rate	2/steam generator	
9.	RCS Subcooled Margin Monitor	1	
10.	PORV/Safety Valve Acoustic Flow Monitoring	1/valve	
11.	PORV Solenoid Power Indication	1/valve	

ATTACHMENT 9

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure	M	R
2.	Wide Range Logarithmic Neutron Flux Monitor	M	N.A.
3.	Reactor Coolant Outlet Temperature	M	R
4.	Pressurizer Pressure	M	R
5.	Pressurizer Level	M	R
6.	Steam Generator Pressure	M	R
7.	Steam Generator Level (Wide Range)	M	R
8.	Auxiliary Feedwater Flow Rate	M	R
9.	RCS Subcooled Margin Monitor	M	R
10.	PORV/Safety Valve Acoustic Monitor	N.A.	R
11.	PORV Solenoid Power Indication	N.A.	N.A.

PLANT SYSTEMS ATTACHMENT 10

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two auxiliary feedwater trains consisting of one steam driven and one motor driven pump and associated flow paths capable of automatically initiating flow shall be OPERABLE. (An OPERABLE Steam driven train shall consist of one pump aligned for automatic flow initiation and one pump aligned in standby[1].) *

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one motor driven pump inoperable:

 1. Restore the inoperable motor driven pump to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With one steam driven pump inoperable: *

 1. Align the standby steam driven pump to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours, and

 2. Restore the inoperable steam driven pump to standby status (or OPERABLE status if the other steam driven pump is to be placed in standby) within the next 30 days or be in HOT SHUTDOWN within the next 12 hours.

c. Whenever a subsystem (consisting of one pump, piping, valves and controls in the direct flow path) required for operability is inoperable for the performance of periodic testing (e.g. manual discharge valve closed for pump Total Dynamic Head test) a dedicated operator will be stationed at the local station with direct communication to the Control Room. Upon completion of any testing, the subsystem required for operability will be returned to its proper status and verified in its proper status by an independent operator check.

 With the requirements of 3.7.1.2.a. or 3.7.1.2.b. met, the provisions of specification 3.0.4 are not applicable. *

---

[1]A standby pump shall be available for operation but aligned so that automatic flow initiation is defeated upon AFAS actuation. *

ATTACHMENT 11

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater flowpath shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each steam driven pump develops a Total Dynamic Head of $\geq$ 2800 ft. on recirculation flow. (If verification must be demonstrated during startup, surveillance testing shall be performed upon achieving an RCS temperature $\geq$ 300°F and prior to entering MODE 1). *

2. Verifying that the motor driven pump develops a Total Dynamic Head of $\geq$ 3100 ft. on recirculation flow.

3. Cycling each testable, remote operated valve that is not in its operating position through at least one complete cycle.

4. Verifying that each valve (manual, power operated or automatic) in the direct flow path is in its correct position. The AFW flow control valves may be verified by observing a 160 gpm setpoint on the flow indicator controller in Control Room.

b. Before entering MODE 3 after a COLD SHUTDOWN of at least 14 days by completing a flow test that verifies the flow path from the condensate storage tank to the steam generators.

c. At least once per 18 months by verifying that each automatic valve in the flow path actuates to its correct position and each auxiliary feedwater pump automatically starts and delivers a modulated flow of 160 gpm $\pm$ 10 gpm to each flow leg upon receipt of each auxiliary feedwater actuation system (AFAS) test signal.

U	=	Maximum number of inoperable safety valves per operating steam line
106.5	=	Power Level-High Trip Setpoint for two loop operation
46.8	=	Power Level-High Trip Setpoint for single loop operation with two reactor coolant pumps operating in the same loop
X	=	Total relieving capacity of all safety valves per steam line in lbs/hour
Y	=	Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A capacity of 400 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the shutdown cooling system may be placed into operation.

Flow control valves, installed in each leg supplying the steam generators, maintain a nominal flow setpoint of 160 gpm plus or minus 10 gpm for operator setting band. The nominal flow setpoint of 160 gpm incorporates a total instrument loop error band of plus 47 gpm (217 gpm total flow per leg) and minus 60 gpm (90 gpm total flow per leg). *

In the spectrum of events analyzed in which automatic initiation of auxiliary feedwater occurs the nominal setting of 160 gpm allows a minimum of 10 minutes before operator action is required. At 10 minutes after automatic initiation of flow the operator is assumed to be available to increase or decrease auxiliary feedwater flow to that required for existing plant conditions.

5.0 Initial Startup Test Program for Auxiliary Feedwater Actuation System

5.1 Background Discussion

Cycle 5 and subsequent cycles will incorporate long term Auxiliary Feedwater System upgrade requirements discussed in NUREG-0737 consisting of a new electric driven train and associated controls. During the startup of Unit 1, Cycle 6 difficulty was experienced in maintaining adequate steam generator pressures during testing of the auxiliary feedwater automatic initiation logic. This was due to the lack of an adequate heat source while feeding the generators in MODE 3 following refueling. In order to perform an adequate test nuclear heat is required to maintain steam generator pressure constant during the the short period of time necessary to complete testing and adjustment of the feedwater system.

5.2 Technical Specification Exemption

In order to use nuclear heat a one-time exception to Technical Specification 3.7.1.2 is required to allow entry into MODE 1 with a partially surveilled Auxiliary Feedwater System. Prior to entry into MODE 2 the following measures will be taken to insure operability of the Auxiliary Feedwater System: 1) system cleanliness will be verified; 2) system hydrostatic tests (within the scope requested in Reference 1) will be performed; 3) electric pump and control valve checks will be performed; and 4) actuation logic control checks will be made. Additionally, surveillance on the steam driven Auxiliary Feedwater Pumps (AFP) will be performed to prove the pumps are available for manual steam generator level control prior to entry into MODE 2. The remainder of testing to be performed during MODE 1 will consist of steam and electric driven pumps automatic flow initiation performance and equipment response time testing at normal operating temperature and pressure.

Once MODE 1 is entered for testing any unrelated problem which results in an exit from MODE 1 will not preclude reentering MODE 1 to complete testing.

5.3 Justification

The Auxiliary Feedwater System is designed to provide feedwater to the steam generator for the removal of heat from the Reactor Coolant System and to cool the Reactor Coolant System to 300° in the event of the loss of normal feedwater flow. The Auxiliary Feedwater Automatic Initiation Logic provides an adequate primary heat sink in the unlikely event that no operator action is taken for up to 10 minutes following a low steam generator water level transient.

The system will be hydrostatically tested prior to entering MODE 3 and all system flushes will be completed. Prior to entry into MODE 2 the Auxiliary Feedwater System will be verified operational from the standpoint that the steam driven AFPs are available for manual feed addition. In addition, the flow control valve will be stroke tested to ensure the operator has manual control capability. Upon entering MODE 2, experienced operations personnel will be available to manually operate the system using the steam driven AFPs to maintain adequate steam generator inventory. Should an event occur which requires the mitigating effect of auxiliary feedwater, decay heat will be minimal. The ambient heat losses at the end of refueling nearly equal the production of heat from the reactor plus the heat from the Reactor Coolant Pumps. Decay heat alone is not enough to overcome ambient heat losses.

Since the system will be functionally capable of maintaining steam generator inventory and operations personnel will be available to operate the system, the intent of Technical Specification 3.7.1.2 will be met.

---

1. A. E. Lundvall to R. A. Clark ltr. dated 8/30/82, "Inservice Inspection Program for relief from ASME Code Section II Requirements Determined to be Impracticable."

6.0 References For Supplement

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2. CENPD-188-A, "HERMITE Space-Time Kinetics," July, 1976.

3. CENPD-161-P, TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," July, 1975.

4. R. V. MacBeth, "An Appraisal of Forced Convection Burn-Out Data," Proc. Instn. Mech. Engrs., 1965-66, Vol. 180, Pt. 3C, pp. 37-50.

5. D. M. Lee, "An Experimental Investigation of Forced Convection Burnout in High Pressure Water; Part IV, Large Diameter Tubes at About 1600 Psia," AEEW-R, November, 1966.

6. CENPD-206-P, "TORC Code, Verification and Simplified Modeling Methods," January, 1977.