TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

September 14, 1984

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

Please refer to TVA's letter of June 19, 1984 which transmitted various comments/proposed modifications to the proof and review version of the Watts Bar Nuclear Plant (WBN) unit 1 Appendix A Technical Specifications.

Enclosed are additional comments/proposed modifications. The majority of these proposed changes are a result of:

(1) the June 25-29, 1984 NRC-OIE inspection of the draft unit 1 Technical Specifications,

(2) the August 7-8, 1984 TVA/NRC-NRR Technical Specification meeting, and

(3) recent TVA/NRC-NRR conference calls held to discuss issues related to the Technical Specifications.

Several of the enclosed comments/proposed modifications have previously been submitted but are now accompanied with additional bases/justifications in support of the requested change.

If you have any questions concerning this matter, please get in touch with D. B. Ellis at FTS 858-2681.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Licensing

8409240077 840914
PDR ADOCK 05000390
A PDR

Sworn to and subscribed before me
this 14th day of Sept 1984

Bryant M. Lowery
Notary Public
My Commission Expires 4/8/86

Enclosure

cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

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On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The steam/feedwater flow mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The steam/feedwater flow mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.51 x 106 lbs/hour. The steam generator low water level portion of the trip is activated when the water level drops below 10.6% of narrow range span between 0 and 30% load, increasing to 53.5% of narrow range span at 100% of nominal load. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

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15.6

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers

3/4 1-7, 3/4 1-8, 3/4 1-9, 3/4 1-10, 3/4 5-9, and B 3/4 1-2

The cold overpressure analysis established that one and only one charging pump can be operable with the Reactor Coolant System (RCS) temperature less tnan 310{o}F. TVA's submittal of June 19, 1984 included a proposal to implement the tagout of the second pump at a RCS temperature of 350{o}F rather than 310{o}F. The basis for this proposal was to coordinate the pump tagout with mode change between modes 3 and 4. A more detailed discussion of each requested change is provided below.

Specification 3.1.2.4 requires that two charging pumps be operable in mode 3 and above. The footnoe associated with this specification should be deleted because it is inconsistent with the APPLICABILITY Statement.

Specification 3.1.2.2 requires that at least two boron injection flow paths be operable with the RCS temperature greater than 310{o}F. TVA proposes that mode 4 be deleted from the applicability, that the ACTION statement refer to HOT SHUTDOWN rather than COLD SHUTDOWN, and that the associated footnote be deleted. These changes will permit TVA to coordinate the charging pump tagout with the mode change between modes 3 and 4. These changes will result in only one boron injection flow path between 310{o}F and 350{o}F. Only one flow path is permitted with a RCS temperature less than 310{o}F. These changes will increase, on average, the period of time which only one flow path will exists by one to two per startup or shutdown.

Specification 3.1.2.1 should have the APPLICABILITY statement revised to include mode 4. This will cover the area deleted by the change to Specification 3.1.2.2 as discussed above.

Specification 3.5.3 should have the associated footnote revise to reference 350{o}F to be consistent with the changes noted above.

Bases 3/4.1.2 should be revised to require single failure proof redundancy when the RCS temperature is greater than 350{o}F and to explain the coordination between tagout and mode change.

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3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.
4,

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 145°F when a flow path from the boric acid tanks is used; and

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 in position, is in its correct position.

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LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two* of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System; and

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore ~~at least two~~ flow paths to OPERABLE status within the next 7 days or be in ~~COLD~~ HOT SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 145°F when it is a required water source;

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 in position, is in its correct position;

c. 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 test signal; and

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 10 gpm to the Reactor Coolant System.

~~*Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 310°F.~~

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CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two# charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in ~~COLD~~ HOT SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying that a differential pressure across each pump of greater than or equal to 2400 psid is developed when tested pursuant to Specification 4.0.5.

~~#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F.~~

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3/4.5.3 ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the RHR heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one centrifugal charging pump and one Safety Injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to ~~310~~°F.
350°

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the P-12 interlock is above its Setpoint, (4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (5) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above ~~200~~ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6542 gallons of 20,000-ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000-ppm borated water from the refueling water storage tank.

With the RCS temperature below ~~200~~ 350°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 310°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. TVA has elected to use a temperature of 350°F to coordinate pump tagout with the mode change.

3/4 1-9, 3/4 1-10, 3/4 5-7 and 3/4 5-10

Surveillance Requirements 4.1.2.3.2 and 4.5.3.2 should be revised as indicated to clarify the methods that could be used to place a pump out of service.

Surveillance Requirements 4.1.2.3.1, 4.1.2.4 and 4.5.2.f should be revised as indicated to specify the correct unit measurement of pressure (i.e., discharge pressure, psig). The accident analyses assumed a discharge pressure.

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LIMITING CONDITION FOR OPERATION

3.1.2.3 One and only one charging pump 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 4, 5, and 6.

ACTION:

With no charging pump 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 shall be demonstrated OPERABLE by verifying that a ~~differential~~ discharge pressure across the pump of greater than or equal to 2400 psig is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, excluding ~~the~~ above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 ~~days, except~~ when the reactor vessel head is removed, by verifying that the ~~motor circuit breakers are secured in the open position.~~ pumps are in the pull-to-lock position and/or the motor circuit breakers are tagged out.

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LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two# charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying that a ~~differential~~ pressure across each pump of greater than or equal to 2400 ~~psid~~ psig is developed when tested pursuant to Specification 4.0.5. discharge

#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F.

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SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation and Automatic Switchover to Containment Sump test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump ≥ 2400 ~~pcid~~ psig,

2) Safety Injection pump ≥ 1407 ~~pcid~~ psig (recirculation flow), and

3) RHR pump ≥ 165 ~~pcid~~ psig (recirculation flow).

g. By verifying the correct position of the following ECCS throttle valves:

1) Within 4 hours following completion of each valve movement or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

CC Discharge Valve Number	SI Cold Leg Throttle Valves Valve Number	SI Hot Leg Throttle Valves Valve Number
63-582	63-550	63-542
63-583	63-552	63-544
63-584	63-554	63-546
63-585	63-556	63-548

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SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All charging pumps and Safety Injection pumps, ~~except the~~ above required OPERABLE pumps, shall be demonstrated ~~inoperable~~ at least once per 12 hours ~~whenever the temperature of one or~~ more of the RCS cold legs ~~is~~ less than or equal to 310°F by verifying that the ~~motor circuit breakers are secured in the open position~~ pumps are in the pull-to-lock position and/or the motor circuit breakers are tagged out.

3/4 2-1

Specification 3.2.1 - correct the typographical error (i.e., based should be band).

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about the target flux difference when THERMAL POWER is above 50% of RATED THERMAL POWER. The indicated AFD may deviate outside the above required target ~~based~~ band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the ±5% target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux# - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

*See Special Test Exception 3.10.2.

#Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

3/4 3-23, 3/4 3-31, 3/4 3-41 and B 3/4 3-2

Tables 3.3-3, 3.3-4 and 4.3-2 and Bases 3/4.3.1 and 3/4.3.2 should be revised as indicated to incorporate P-14 interlock.

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
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	19
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	19
c. Reactor Trip, P-4	2	2	2	1, 2, 3	21
d. Steam Generator Level, P-14	3/stm.gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. in each operating stm. gen.	1, 2	15*

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. 6.9 kV Shutdown Board		
a. Loss of Power		
1) Start Diesel Generator	0.0 volts with a 1.5 second time delay	0.0 volts with a 1.5 ± 0.5 second time delay
2) Load Shedding	0.0 volts with a 5 second time delay	0.0 volts with 5 ± 1 second time delay
b. Degraded Voltage		
1) Voltage Sensor	6560 v	6560 ± 33 v
2) Diesel Generator Start and Load Shedding Timer	300 s	300 ± 30 s
3) Safety Injection Degraded Voltage Logic Enable Timer	10 s	9.5 ± 0.5 s
9. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	≤ 1970 psig	≤ 1980 psig
b. Low-Low T_{avg}, P-12, increasing	≥ 550°F	≤ 552°F
decreasing	≤ 550°F	≥ 548°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Level, P-14	≤ 82.4 % of narrow range instrument span each steam generator	≤ 82.4 % of narrow range instrument span each steam generator

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Level, P-14	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2

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

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped, blocks steam dump via load rejection controller and arms steam dump via plant trip controller.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure.

P-12 On increasing reactor coolant loop temperature, P-12 automatically reinstates Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure, and provides an arming signal to the steam dump system. On decreasing reactor coolant loop temperature, P-12 allows the manual block of Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure and automatically removes the arming signal from the Steam Dump System.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

P-14 On increasing steam generator levels, P-14 automatically trips all feedwater pumps, initiates a turbine trip, closes the feedwater isolation valves, and inhibits feedwater control valve modulation. On decreasing steam generator levels, P-14 allows the start of all feedwater pumps, permits turbine operation, allows feedwater isolation valve openning, and allows feedwater control valve modulation.

3/4 3-30 and 3/4 3-31

Table 3.3-4 items 6.e and 8.a should be revised as indicated to clarify nominal voltage setpoints and relay response times.

Table 3.3-4 item 7.b should be revised to specify an allowable value of 32.5 inches.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. Auxiliary Feedwater (continued)		
e. ~~Loss-of-Offsite~~ Power-Start Motor-Driven Pumps	4860 volts	4860 ± 97.2 volts
Start Turbine-Driven Pump	0.0 volts with a 5.0 second time delay	0.0 volts with a 5.0 ± 1.0 second time delay
1) Nominal Voltage Setpoint 2) Relay Response Time		
f. Trip of All Main Feedwater ~~Pumps - Start~~ Motor-Driven Pumps and ~~Turbine-~~Driven Pump	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low (Suction Transfer to ERCW)		
1) Supply Valve for Motor-Driven Pump	≥ 1.7 psig	≥ 1.7 ± 0.5 psig
2) Supply Valve for Turbine-Driven Pump	≥ 11.1 psig	≥ 11.1 ± 0.5 psig
7. Automatic Switchover To Containment Sump		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. RWST Level - Low Coincident With	≥ 130" from tank base	≥ 126" from tank base
Containment Sump Level - High	≤ 30" above elev. 703'	≤ 32.5" ~~≤ 30" ± 2.5"~~ above elev. 703'
And		
Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints/ Allowable Values	

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. 6.9 kV Shutdown Board		
a. Loss of ~~Power~~ Voltage	4860 volts	4860 ± 97.2 volts
1) Start Diesel Generator	0.0 volts with a	0.0 volts with a
a. Nominal Voltage Setpoint	1.5 second time delay	1.5 ± 0.5 second time delay
b. Relay Response Time	4860 volts	4860 ± 97.2 volts
2) Load Shedding	0.0 volts with a	0.0 volts with
a. Nominal Voltage Setpoint	~~5~~ second time delay	~~5~~ ± ~~1~~ second time delay
b. Relay Response Time	5.0	5.0 1.0
b. Degraded Voltage		
1) Voltage Sensor	6560 v	6560 ± 33 v
2) Diesel Generator Start and Load Shedding Timer	300 s	300 ± 30 s
3) Safety Injection Degraded Voltage Logic Enable Timer	10 s	9.5 ± 0.5 s
9. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	≤ 1970 psig	≤ 1980 psig
b. Low-Low T_{avg}, P-12, increasing	≥ 550°F	≤ 552°F
decreasing	≤ 550°F	≥ 548°F
c. Reactor Trip, P-4	N.A.	N.A.

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3/4 3-44, 3/4 3-45, 3/4 3-46, 3/4 3-47, 3/4 4-18, B 3/4 3-2 and B 3/4 4-4

Tables 3.3-6 and 4.3-3 should be revised as indicated to improve clarity and elminate redundancy. Monitor indentifiers have been added.

The requirements for containment purge isolation are addressed in techncial specification 3.3.3.10, item 7 of Tables 3.3-13 and 4.3-9 and are therefore redundant and should be deleted from Tables 3.3-6 and 4.3-3.

The fuel storage pool area has two fully redundant monitors which perform both radiation and criticality detection; therefore, listing separate requirements and setpoints for the same monitor is confusing and should be deleted from Tables 3.3-6 and 4.3-3.

The containment atmosphere monitor requirements should be deleted from Tables 3.3-6 and 4.3-3 since they are redundnat to technical specification 3.4.6.1. The minimum channels operable, applicable modes, and ACTION requirements are identical; however, the surveillance frequencies should be included in Surveillance Requirement 4.4.6.1 as indicated.

The setpoint in Table 3.3-6 for main control room isolation should be specified in units used on the monitor and a correlation between the setpoint and 10 CFR 20 limit should be sited.

ACTION statement 28 for Table 3.3-6 should be revised as indicated to identify the correlation between the subject monitors and the Auxiliary Building Gas Treatment System.

Bases 3/4.3.3.1 and 3/4.4.6.1 should be revised as indicated for clarification purposes.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
~~1. Containment Purge Isolation~~					
~~a. Containment Atmosphere Radioactivity High~~	~~1~~	~~1~~	~~1, 2, 3, 4~~	~~≤ 100 mR/h~~	~~26~~
~~b. Containment Purge Exhaust Radioactivity High~~	~~1~~	~~1~~	~~1, 2, 3, 4~~	~~#~~	~~26~~
1. ~~2.~~ Fuel Storage Pool Area-Ventilation Isolation (RE-90-102 and RE-90-103)					
~~a. Spent Fuel Pool Radioactivity High~~	1	2	**	≤ ~~50~~ 15 mR/h	28
~~b. Criticality Radiation Level~~	~~1~~	~~2~~	~~*~~	~~≤ 15 mR/h~~	~~28~~
~~3. Containment Atmosphere~~					
~~a. Gaseous Radioactivity RCS Leakage Detection~~	~~N.A.~~	~~1~~	~~1, 2, 3, 4~~	~~N.A.~~	~~29~~
~~b. Particulate Radioactivity RCS Leakage Detection~~	~~N.A.~~	~~1~~	~~1, 2, 3, 4~~	~~N.A.~~	~~29~~

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TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
2. ~~A.~~ Control Room Ventilation Isolation (RE-90-125 and RE-90-126)					
Control Room Air Intake Radioactivity-High	1	2	All	≤ ~~2mR/h~~ 400 cpm*	27

TABLE NOTATIONS

*~~With fuel in the fuel storage areas~~. 400 cpm is equivalent to 1×10^{-5} uCi/cc of Xe-133. The setpoint is considered to be above naturally occurring background radiation.

**With irradiated fuel in the fuel storage areas.

~~#Must satisfy Specification 3.3.3.10 requirements~~.

ACTION STATEMENTS

~~ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge valves are maintained closed.~~

ACTION 27 - With the number of operable channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Ventilation System and initjate operation of the Control Room Ventilation System in the recirculation mode.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel pool area. ~~Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool area.~~ and satisfy the ACTION requirements for Specification 3.9.12.

~~ACTION 29 - Must satisfy the ACTION requirement for Specification 3.4.6.1.~~

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
~~1. Containment Purge Isolation~~				
~~a. Containment Atmosphere Radioactivity High~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
~~b. Containment Purge Exhaust Radioactivity High~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
1. ~~2.~~ Fuel Storage Pool Area-Ventilation Isolation (RE-90-102 and RE-90-103)				
~~a. Spent Fuel Pool Radioactivity High~~	S	R	M	**
~~b. Criticality High Radiation Level~~	~~S~~	~~R~~	~~M~~	~~*~~
~~3. Containment Atmosphere~~				
~~a. Gaseous Radioactivity RCS Leakage Detection~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
~~b. Particulate Radioactivity RCS Leakage Detection~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
~~1. Containment Purge Isolation~~				
~~a. Containment Atmosphere Radioactivity High~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
~~b. Containment Purge Exhaust Radioactivity High~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
1. ~~2.~~ Fuel Storage Pool Area-Ventilation Isolation (RE-90-102 and RE-90-103)				
~~a. Spent Fuel Pool Radioactivity High~~	S	R	M	**
~~b. Criticality High Radiation Level~~	~~S~~	~~R~~	~~M~~	~~*~~
~~3. Containment Atmosphere~~				
~~a. Gaseous Radioactivity RCS Leakage Detection~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
~~b. Particulate Radioactivity RCS Leakage Detection~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~

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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 Lower Containment Atmosphere Gaseous Radioactivity Monitoring System,

b. The Containment Pocket Sump Level Monitoring System, and

c. The Lower Containment Atmosphere Particulate Radioactivity Monitoring System.

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

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Lower Containment Atmosphere Gaseous And ~~Particulate~~ Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST ~~at the frequencies specified in Table 4.3-3, and~~

b. Containment Pocket Sump Level Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

at least once per 12 hours, once every 31 days and once every 18 months, respectively, and

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

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped, blocks steam dump via load rejection controller and arms steam dump via plant trip controller.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure.

P-12 On increasing reactor coolant loop temperature, P-12 automatically reinstates Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure, and provides an arming signal to the steam dump system. On decreasing reactor coolant loop temperature, P-12 allows the manual block of Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure and automatically removes the arming signal from the Steam Dump System.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems. The radiation monitor setpoints listed in the Specification are assumed to be setpoints above local background radiation.

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STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. The radiation monitor requirements can be satisfied by RE-90-106 or RE-90-112 provided the monitor is sampling the lower compartment.

~~3/4.4.6.2~~ OPERATIONAL ~~LEAKAGE~~

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 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 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 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 CONTROLLED LEAKAGE limitation restricts opeation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.

Specification 4.3.3.3.2 should be revised as indicated to require that actuated seismic monitoring equipment be restored to OPERABLE status regardless of the level of seismic activity.

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INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

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

4.3.3.3.2 Each of the above required seismic monitoring instruments actuated during a seismic event ~~greater than or equal to 0.01 g~~ shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

3/4 3-56

Table 3.3-9 should be revised as indicated to correctly list the total number of remote shutdown monitoring instruments.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Source Range Nuclear Flux	Note 1	1	1
2. Reactor Trip Breaker Indication	At trip switchgear	1/trip breaker	1/trip breaker
3. Reactor Coolant Temperature - Hot Leg	Note 1	1/loop	1
4. Pressurizer Pressure (wide Range)	Note 1	1	1
5. Pressurizer Level	Note 1	~~1~~ 2	1
6. Steam Generator Pressure	Note 1	1/stm gen	1/steam generator
7. Steam Generator Level	Note 2	1/stm gen	1/steam generator
8. Control Rod Bottom Bistable	R-41-R-44	1 Rod Bottom Bistable/rod	1 Rod Bottom Bistable/rod
9. RHR Flow Rate	Note 1	1/train	1
10. RHR Temperature	Note 1	1/train	1
11. Auxiliary Feedwater Flow Rate	Note 1	1/stm gen	1
12. Pressurizer Relief Tank Pressure	Note 1	1	1
13. Containment Pressure	Note 1	1	1

Note 1: Auxiliary Control Room Panel 1-L-10
Note 2: Auxiliary Control Room Panel 1-L-10 or near Auxiliary Feedwater Pump

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Specification 4.3.3.8.1 should be revised and proposed Specification 4.3.3.8.3 should be incorporated as indicated.

The National Fire Protection Association (NFPA) code specifies different test intervals for smoke and infrared or thermal detectors. The infrared and thermal detectors are not subject to failures due to clogging as are the smoke detectors. Less frequent testing is justified on this basis.

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INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instrument shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the location listed in Specification 4.6.1.6.

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instrument shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6.

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

SURVEILLANCE REQUIREMENTS

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

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

Insert ~~attached~~ SR 4.3.3.8.3

Proposed SR 4.3.3.8.3

4.3.3.8.3 One of the above required infrared or thermal detection instruments in each zone which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST on one of the above required infrared or thermal detection instruments in each zone during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months. Detectors will be selected such that all infrared or thermal detectors are tested.

3/4 3-64

This proposed modification to Specification 3.3.3.8 addresses the NRC OIE concern of automatic actuation system inoperability due to fire detector zone inoperability.

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INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instrument shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the location listed in Specification 4.6.1.~~8~~.5

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instrument shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.~~8~~.5

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

SURVEILLANCE REQUIREMENTS

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

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

c. With an automatic suppression system inoperable due to the inoperability of all the function B detectors within a given zone, comply with the actions listed in Specification 3.7.11.2 or 3.7.11.3, as applicable, unless the zone is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperatures at least once per 1 hour at the locations listed in Specification 3.6.1.5.

3/4 3-67 and 3/4 3-69

Table 3.3-11 should be revised as indicated to incorporate minor editorial changes and additions to the Fire Detection Instrument list.

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTATION

ZONE	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS** HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
B.	Control Building (Continued)			
218	CR Fltr. B, El. 755			0/1
219	CR Fltr. B, El. 755			0/1
220	Main CR, El. 755			27/0
226	Electric Cont. Bds., El. 755			12/0
229	Main Cont. Bds., El. 755			8/0
221	Tech Support Center, El. 755			0/6
222	Tech Support Center, El. 755			0/6
223	PSO Eng. Shop, El. 755			0/1
224	PSO Eng. Shop, El. 755			0/1
225	Relay Bd. Rm., El. 755			0/12
267	Aux. Instr. Rm., Unit 1, El. 708			0/8
268	Aux. Instr. Rm., Unit 1, El. 708			0/10
271	Aux. Instr. Rm., Unit 2, El. 708			0/8
272	Aux. Instr. Rm., Unit 2, El. 708			0/10
298	Common Main Cont. Boards & M-15			12/0
412	Duplex Relay Bds., El. 755			4/0
50	Mech. Equip. Rm. Col. C1, El. 692			0/2
51	Mech. Equip. Rm. Col. C1, El. 692			0/2
52	Mech. Equip. Rm. Col. C3, El. 692			0/2
53	Mech. Equip. Rm. Col. C3, El. 692			0/2
387	Control/Turbine Bldg. Wall	0/24		
C.	Auxiliary Building			
39	Cont. Spray Pump 1A-A, El. 676			2/0
40	Cont. Spray Pump 1B-B, El. 676			2/0

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTATION

ZONE	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS** HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
C.	Auxiliary Building (Continued)			
98	Cntmt. Purge Air Fltr., El. 713			0/2
99	Cntmt. Purge Air Fltr., El. 713			0/2
102	Pipe Gallery, El. 713			0/4
103	Pipe Gallery, El. 713			0/4
106	Aux. Bldg. A5-A11, Col. T-W, El. 713			0/8
107	Aux. Bldg. A5-A11, Col. T-W, El. 713			0/8
110	Aux. Bldg. A1-A8, Col. Q-U, El. 713			0/12
111	Aux. Bldg. A1-A8, Col. Q-U, El. 713			0/12
112	Aux. Bldg. A8-A15, Col. Q-U, El. 713			0/9
113	Aux. Bldg. A8-A15, Col. Q-U, El. 713			0/9
114	Waste Packaging Area, El. 729			0/3
115	Waste Packaging Area, El. 729			0/3
120	Aux. Bldg. Gas Trtmt. Fltr., El. 729			0/1
121	Aux. Bldg. Gas Trtmt. Fltr., El. 729			0/1
123	Vol. Control Tank Rm. 1A, El. 713			0/1
125	Vol. Control Tank Rm. 1A, El. 713			0/1
128	Post Accident Sample Rm. U1, ~~Fuel Transfer Valve Rm.~~, El. 729			0/~~2~~ 3
129	Post Accident Sample Rm. U1, ~~Fuel Transfer Valve Rm.~~, El. 729			0/~~2~~ 3
132	Ventilation & Purge Air Rm., El. 737			0/~~3~~ 5
133	Ventilation & Purge Air Rm., El. 737			0/~~3~~ 5
134	Aux. Bldg. A5-A11, Col. U-W, El. 737			0/7
135	Aux. Bldg. A5-A11, Col. U-W, El. 737			0/7
136	Heating & Vent Rm., El. 737			0/4
137	Heating & Vent Rm., El. 737			0/4

Table 3.3-13 ACTION Statement 39 should be revised as indicated to better define the type of radioactivity analysis.

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

*** During operation of the Containment Purge System, Auxiliary Building Gas Treatment System, or waste gas decay tank disposal.

**** At all times other than when the most recent Secondary Coolant System specific activity sample and analysis program gross radioactivity determination is less than or equal to 1×10^{-6} uCi/gm.

ACTION STATEMENTS

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for ~~radioactivity~~ within 24 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this Waste Gas Disposal System may continue for up to 7 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours to meet the requirements of Specification 3.11.2.5. With the hydrogen and oxygen monitors inoperable, be in at least HOT STANDBY within 6 hours.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided that samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING or VENTING of radioactive effluents via this pathway.

noble gas gross radioactivity or an isotopic analysis is performed with LLD's as given in Table 4.11-2

TVA's submittal of June 19, 1984 included a request to revise ACTION statement 40 of Table 3.3-13 with respect to the inoperability of the H_2 and O_2 monitors.

Several conference calls have recently been held between TVA and NRC representativs to discuss this proposal. As a result of these discussions, TVA now proposes to address the inoperability of the H_2 and O_2 monitors by submittal of a Special Report.

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TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

*** During operation of the Containment Purge System, Auxiliary Building Gas Treatment System, or waste gas decay tank disposal.

**** At all times other than when the most recent Secondary Coolant System specific activity sample and analysis program gross radioactivity determination is less than or equal to 1×10^{-6} uCi/gm.

ACTION STATEMENTS

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 40 - ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this Waste Gas Disposal System may continue for up to 7 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours to meet the requirements of Specification 3.11.2.5. With the hydrogen and oxygen monitors inoperable, be in at least HOT STANDBY within 6 hours.~~

Insert proposed ACTION 40

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided that samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING or VENTING of radioactive effluents via this pathway.

Proposed ACTION 40 Table 3.3-13

ACTION 40 - With the number of channels OPERABLE less than reuired by the Minimum Channels OPERABLE requirement, operation of this Waste Gas Disposal System may continue provided grab samples are collected at least once per batch transfer to a waste gas decay tank and at least once per 4 hours and analyzed within the followign 4 hours to meet the requirements of Specification 3.11.2.5. With either the hydrogen or oxygen monitor inoperable for more than 7 days or with both oxygen and hydrogen monitors inoperable, prepare and submit to the Commission with 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for the inoperability, action(s) taken to restore the monitor(s) to OPERABLE status, and a summary description of action(s) taken to prevent recurrence.

Specification 3.4.3 should be revised as indicated to emphasize the indicated level measurement that the operator will use.

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3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with ~~a water volume~~ an indicated level of less than or ~~equal to 1656 cubic feet equivalent to an indicated level of less than or~~ equal to 92% on narrow range indication (equivalent to a water volume of less than or equal to 1656 cubic feet), and at least two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply.

3/4 4-20

Surveillance Requirement 4.4.6.2.1.d should be revised as indicated to clarify the fact that several alarms can indicate intersystem leakage. A description of the alarms used for this purpose is provided in Section 5.2.5 of the Watts Bar SER (NUREG-0847). TVA believes that this clarification is necessary for proper interpretation of the requirement.

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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 lower containment atmosphere gaseous or particulate radioactivity monitor for relative change at least once per 12 hours;

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

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of ~~a Reactor Coolant~~ System water inventory balance at least once per 72 hours during steady-state operation or within 1 hour of receiving an alarm which indicates intersystem leakage ~~alarm~~; and

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

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months;

b. 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;

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve; and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

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

3/4 4-36 and B 3/4 4-15

Figure 3.4-4 has been revised to be consistent with TVA's submittal of Westinghouse report "Setpoint Program Determination for the Westinghouse Cold Overpressure Mitigating System in the Watts Bar Unit 1 Plant" by letter dated June 7, 1984.

Bases 3/4.4.9 should be revised as indicated to reflect the fact that a cold leg measured temperature is used to protect the reactor pressure from a predicted pressure-temperature condition.

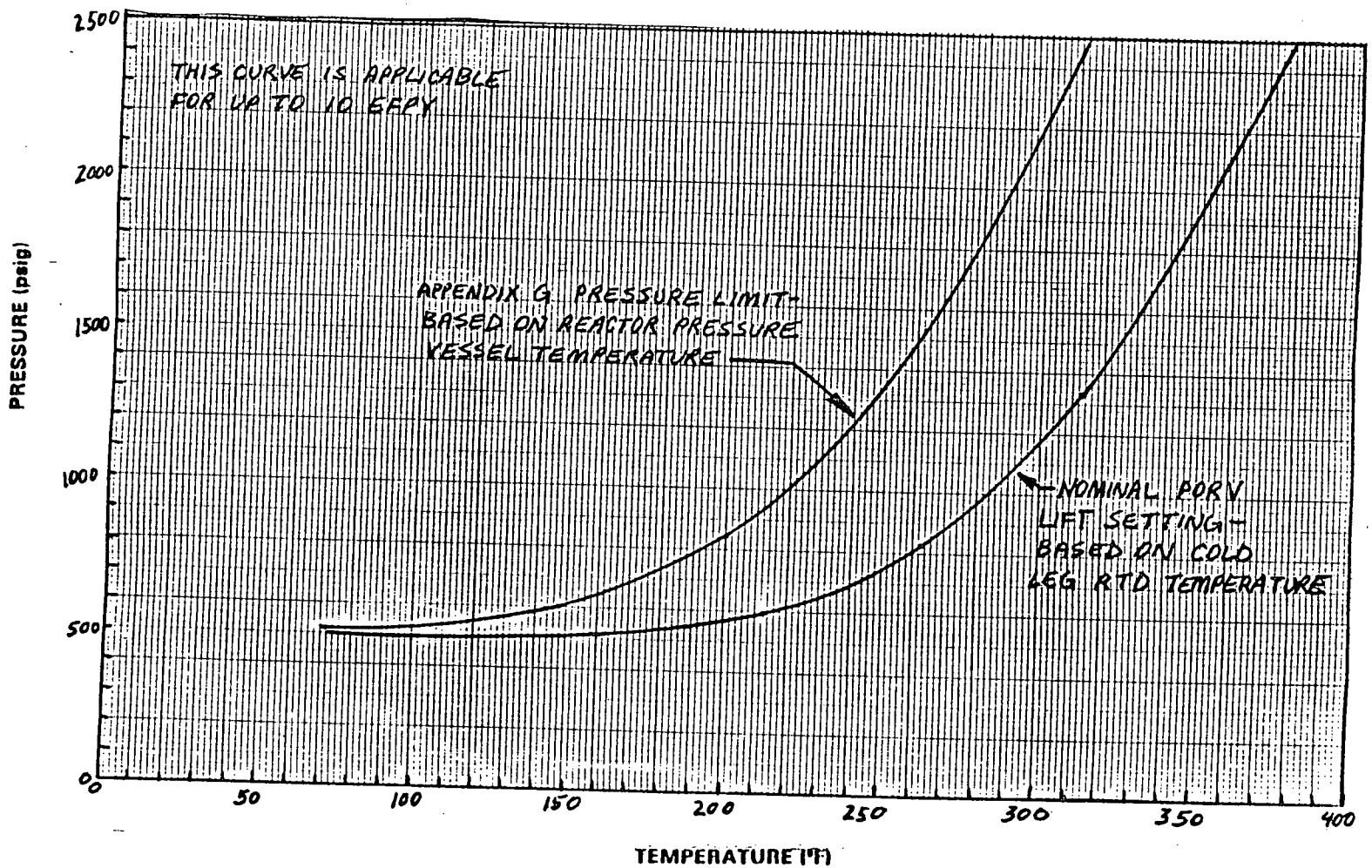

FIGURE 3.4-4

PORV NOMINAL LIFT SETTINGS

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PRESSURE/TEMPERATURE LIMITS (Continued)

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.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

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.

(heat input)

The OPERABILITY of two PORVs or an RCS vent opening of at least 3 square inches ensures that the RCS will be protected from pressure 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 310°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle reactor coolant pump with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a HPSI pump and its injection into a water solid RCS.

(mass input)

Insert proposed paragraph

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 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition and Addenda through Winter 1971.

Proposed Paragraph for page B 3/4 4-15

In developing the PORV setpoint for a given measured RTD temperature, account must be taken of the heat transport effect. If a heat input event were to occur, the cold leg temperature would rapidly rise to that corresponding to the steam generator, while the vessel would still be at the RCS temperature which existed prior to the transient. Therefore, the PORV setpoint must be defined so that it corresponds to the Appendix G limit at the vessel temperature, not the measured RTD temperature. It was assumed that the RTD was measuring a temperature 63°F higher than the vessel (50°F due to primary to secondary temperature difference plus 13°F instrument error). The actural PORV setpoints will be staggered and set below the nominal lift setting shown in Figure 3.4-4.

Surveillance Requirement 4.5.1.2.c.1 should be revised as indicated. The requiement to test the upper head injection isolation valves should be revised to account for the addition of the manual signal. The proposed wording will require only one valve closure. The second signal would be tested upper a common point. This testing scheme is similar to reactor protection system testing. Additional valve closures can lead to valve seat damage due to the high closing velocity.

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SURVEILLANCE REQUIREMENTS

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the solution in the water-filled accumulator,

c. At least once per 18 months by:

1) Verifying that each accumulator isolation valve ~~closes~~ can be closed:

a) Manually from the control room, and

b) Automatically when the water level in the water-filled accumulator is 103.4 ± 0.5 inches above the tank vendor working line.

2) Verifying that the total dissolved nitrogen and air in the water-filled accumulator is less than 80 scf per 1800 cubic feet of water (equivalent to 5×10^{-5} pounds nitrogen per pounds water).

d. At least once per 5 years and if the requirements of Specification 4.5.1.2c.2) are not met by replacing the membrane installed between the water-filled and nitrogen-bearing accumulators.

3/4 5-8

TVA's submittal of June 19, 1984 inadvertently indicated that the centrifugal charging pump flow rate of Surveillance Requirement 4.5.2.h.1.b should be revised to 567 gpm. The correct value is 568 gpm.

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SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 346 gpm, and

b) The total pump flow rate is less than or equal to ~~595~~ 568 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 462 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3976 gpm.

3/4 6-1

Surveillance Requirement 4.6.1.1.b should be revised as indicated for clarification purposes.

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in ~~Table 3.6-2~~ of Specification 3.6.3;

b. By verifying that each containment air lock is OPERABLE by ~~in compliance with~~ the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 15 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

---

*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus and containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

Specification 4.6.1.2 should be revised as indicated to remove the reference to ANSI N45.4-1972. TVA should have the flexibility to use any test method that meets the criteria specified in Appendix J to 10 CFR Part 50.

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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 L_a, 0.25% by weight of the containment air per 24 hours at P_a, 15 psig;

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

c. A combined bypass leakage rate of less than 0.25 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With: (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a, or (b) the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) the combined bypass leakage rate exceeding 0.25 L_a, restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.25 L_a 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~~.

3/4 6-8

Surveillance Requiement 4.6.1.3.a should be revised as indicated. TVA has plans to install a continuous leakage monitoring system on the airlock door seals. The continuous monitoring system can be used in lieu of the individual test. The proposed wording allows for use of either method.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Except for air locks using continuous Leakage Monitoring Systems, Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by pressurizing the volume between the door seals to at least 6 psig for at least 30 seconds and verifying the leakage does not exceed 0.01 L_a;

b. By conducting overall air lock leakage tests at not less than P_a, 15 psig, and verifying the overall air lock leakage rate is within its limit:

1) At least once per 6 months,# and

2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.
*This represents an exemption to Appendix J of 10 CFR Part 50.

3/4 6-14

Surveillance Requirement 4.6.1.8.d.4 should be revised as indicated to address annulus inleakage.

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. 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 0.2%,

d. At least once per 18 months, by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches Water Gauge while operating the system at a flow rate of 4000 cfm ± 10%,

2) Verifying that the system starts automatically on a Phase "A" Isolation test signal,

3) Verifying that the filter cooling bypass valves can be opened,

4) Verifying that the air cleanup subsystem maintains the annulus building ~~at a pressure equal to or~~ more ~~negative~~ than minus 0.5 inches water gage relative to the Shutdown Board Room, ~~and~~ with an inleakage of ≤ 250 cfm, and

5) Verifying ~~that the~~ heaters ~~dissipate~~ 20 ~~± 2.0~~ kW when tested in accordance with ANSI N510-1975.

e. 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 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

f. 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 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

3/4 6-15

By letter dated July 27, 1983, TVA requested that the allowable purge and vent time (Specification 3.6.1.9) for Watts Bar be increased to 2,000 hours per year. The basis for the request was the experience at Sequoyah Nuclear Plant that an average of 5.33 hours of purge and vent time was needed per day to maintain containment internal pressure and reduce and maintain radiation levels. TVA again requested the increase in allowable purge and vent time by letter dated June 19, 1984.

The design of Watts Bar is essentially identical to Sequoyah relative to the items that lead to purging and venting. TVA has reasonably concluded that the purging and venting needs at Watts Bar will be approximately 2,000 hours per year.

10 CFR Part 50.91 states in paragraph 5 that:

> "The Commission expects its licensees to apply for license amendments in a timely fashion. It will decline to dispense with notice and comment on the determination of no significant hazard consideration, if it determines that the licensee has failed to make a timely application for the amendment in order to create the emergency to take advantage of the emergency provision. Whenever a threatened closure or derating is involved, a licensee requesting an amendment must explain why this emergency situation occurred and why it could not avoid this situation, and the Commission will assess the licensee's reasons for failure to file an application sufficiently in advance of that event."

In accordance with 10 CFR Part 50.91, TVA must again request that the allowable purge time for Watts Bar be increased to 2,000 hours per year. Our estimate of purge and vent needs, based on Sequoyah Nuclear Plant experience, leads us to request this time. The present arbitrary time limit of 500 hours per year could lead us to request an emergency Technical Specification change 90 days after operations begins. Our experience with the notice for public comment required by Federal law is that at least 90 days, and typically 180 days, are required to request and receive a routine technical specification change. Therefore, TVA must receive an increase in allowable purge time to avoid the potential for an emergency Technical Specification change request.

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CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 Each containment purge supply and/or exhaust isolation valve shall be OPERABLE and the 24-inch at 70° open, the 8-inch and the 12-inch containment purge supply and/or exhaust isolation valve(s) may be opened for up to ~~500~~ 2000 hours during a calendar year provided no more than one line is open at one time.

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

ACTION:

a. With the 24-inch at 70° open, the 8-inch and the 12-inch containment purge supply and/or exhaust isolation valve(s) open for more than ~~500~~ 2000 hours during a calendar year, close any open containment purge and/or exhaust isolaton valve or isolate the penetration(s) within ~~4 hours~~, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.9.3 and/or 4.6.1.9.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 Each 24-inch containment purge supply and/or exhaust isolation valve(s) shall be verified to be physically restricted to 70° open at least once per 31 days.

4.6.1.9.2 The cumulative time that all containment purge supply and/or exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.

4.6.1.9.3 At least once per 3 months each containment purge supply and/or exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.01 L_a when pressurized to P_a.

3/4 6-17, 3/4 6-19, 3/4 6-20, 3/4 6-21, 3/4 6-22 and 3/4 6-23

Specification 3.6.3 should be revised to delete the reference to the nonapplicability of Specification 3.0.4.

Table 3.6-2 should be revised as indicated to denote those valves for which Specification 3.0.4 does not apply.

Several additional errors have been identified in Table 3.6-2. Valve FCV-77-20 was listed twice and four valves' isolation times need to be corrected to be consistent with Sequoyah.

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3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-2 shall be OPERABLE with isolation times as shown in Table 3.6-2.

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

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-2 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. ~~Isolate each affected penetration within 4 hours by use of at least~~ one deactivated automatic valve secured in the isolation position, ~~and the provisions of Specification 3.0.4 are not applicable~~, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, ~~and the provisions of Specification 3.0.4 are not applicable~~, or

d. Be ~~in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-2 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation		
FCV-1-7*#	SG Blowdown	10
FCV-1-14*#	SG Blowdown	10
FCV-1-25*#	SG Blowdown	10
FCV-1-32*#	SG Blowdown	10
FCV-1-181*#	SG Blowdown	~~10~~ 15
FCV-1-182*#	SG Blowdown	~~10~~ 15
FCV-1-183*#	SG Blowdown	~~10~~ 15
FCV-1-184*#	SG Blowdown	~~10~~ 15
FCV-30-134	Cont. to Annulus ΔP	10
FCV-30-135	Cont. to Annulus ΔP	10
FCV-31C-305#	CW-Inst Room Clrs	10
FCV-31C-306#	CW-Inst Room Clrs	10
FCV-31C-308#	CW-Inst Room Clrs	10
FCV-31C-309#	CW-Inst Room Clrs	10
FCV-31C-326#	CW-Inst Room Clrs	10
FCV-31C-327#	CW-Inst Room Clrs	10
FCV-31C-329#	CW-Inst Room Clrs	10
FCV-31C-330#	CW-Inst Room Clrs	10
FCV-43-22#	Sample RC Outlet Hdrs	10
FCV-43-23#	Sample RC Outlet Hdrs	10
FCV-43-~~540~~ 54D#	Stm Gen No. 1 Bldn Isol Vlv	10
FCV-43-~~568~~ 56D#	Stm Gen No. 2 Bldn Isol Vlv	10
FCV-43-~~598~~ 59D#	Stm Gen No. 3 Bldn Isol Vlv	10
FCV-43-~~630~~ 63D#	Stm Gen No. 4 Bldn Isol Vlv	10
FCV-43-55#	SG Blow Dn Sample Line	10
FCV-43-58#	SG Blow Dn Sample Line	10
FCV-43-61#	SG Blow Dn Sample Line	10
FCV-43-64#	SG Blow Dn Sample Line	10
FCV-61-96#	Gylcol Inlet to Floor Cooler	30
FCV-61-97#	Gylcol Inlet to Floor Cooler	30
FCV-61-110#	Gylcol Outlet to Floor Cooler	30

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TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation (Cont.)		
FCV-61-122 #	Gylcol Outlet to Floor Cooler	30
FCV-61-191 #	Ice Condenser - Gylcol In	30
FCV-61-192 #	Ice Condenser - Gylcol In	30
FCV-61-193 #	Ice Condenser - Gylcol Out	30
FCV-61-194 #	Ice Condenser - Gylcol Out	30
FCV-62-61	RCP Seals	10
FCV-62-63	RCP Seals	10
FCV-62-72 ##	Letdown Line	10
FCV-62-73 ##	Letdown Line	10
FCV-62-74 ##	Letdown Line	10
FCV-62-76 #	Letdown Line	10
FCV-62-77	Letdown Line	~~10~~ 20
FCV-63-23 #	Accum to Hold Up Tank	10
FCV-63-64 #	WDS N_2 to Accum	10
FCV-63-71 #	Accum to Hold Up Tank	10
FCV-63-84 #	Accum to Hold Up Tank	10
FCV-68-305 #	WDS N_2 to PRT	10
FCV-68-307 #	PRT to Gas Analyzer	10
FCV-68-308 #	PRT to Gas Analyzer	10
FCV-70-85 #	CCS from Excess Lt Dn Hx	10
FCV-70-143 #	CCS to Excess Lt Dn Hx	60
FCV-77-9 #	RCDT Pump Disch	10
FCV-77-10 #	RCDT Pump Disch	10
FCV-77-16 #	RCDT to Gas Analyzer	10
FCV-77-17 #	RCDT to Gas Analyzer	10
FCV-77-18 #	RCDT and PRT to V II	10
FCV-77-19 #	RCDT and PRT to V II	10
FCV-77-20 #	RCDT N_2 Supply	10
FCV-77-127 #	Floor Sump Pump Disch	10
FCV-77-128 #	Floor Sump Pump Disch	10
FCV-81-12 #	Primary Water Makeup	10
FCV-87-7 #	UHI Test Line	10
FCV-87-8 #	UHI Test Line	10
FCV-87-9 #	UHI Test Line	10

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TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation (Cont.)		
FCV-87-10* #	UHI Test Line	10
FCV-87-11* #	UHI Test Line	10
FCV-26-240	Fire Protection Isol.	≤ 20
FCV-26-243	Fire Protection Isol.	≤ 20
~~FCV-77-80~~	~~RCDT N_2 Supply~~	~~≤ 20~~
FCV-43-2 #	Sample Przr Steam Space	≤ 5
FCV-43-3 #	Sample Przr Steam Space	≤ 5
FCV-43-11 #	Sample Przr Liquid	≤ 5
FCV-43-12 #	Sample Przr Liquid	≤ 5
FCV-43-34 #	Accum Sample	≤ 5
FCV-43-35 #	Accum Sample	≤ 5
FCV-43-75 #	Boron Analyzer	≤ 5
FCV-43-77 #	Boron Analyzer	≤ 5
2. Phase "B" Isolation		
FCV-32-80	Train A Control Air Isolation	≤ 10
FCV-32-102	Train B Control Air Isolation	≤ 10
FCV-32-110	Non-Essential Control Air Isolation	≤ 10
FCV-67-83 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-87 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-88 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-91 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-95 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-96 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-99 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-103 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-104 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-107 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-111 #	ERCW - LWR Cmpt Clrs	≤ 60
FCV-67-112 #	ERCW - LWR Cmpt Clrs	≤ 60

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TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
2. Phase "B" Isolation (Cont.)		
FCV-67-130#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-131#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-133#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-134#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-138#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-139#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-141#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-142#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-295#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-296#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-297#	ERCW - Up Cmpt Clrs	≤ 60
FCV-67-298#	ERCW - Up Cmpt Clrs	≤ 60
FCV-70-87	RCP Thermal Barrier Ret	≤ 60
FCV-70-89	CCS from RCP Oil Coolers	≤ 60
FCV-70-90	RCP Thermal Barrier Ret	≤ 60
FCV-70-92	CCS from RCP Oil Coolers	≤ 60
FCV-70-134	To RCP Thermal Barriers	≤ 60
FCV-70-140	CCS to RCP Oil Coolers	≤ 60
3. Phase "A" Containment Vent Isolation		
FCV-30-7#	Upper Compt Purge Air Supply	≤ 4
FCV-30-8#	Upper Compt Purge Air Supply	≤ 4
FCV-30-9#	Upper Compt Purge Air Supply	≤ 4
FCV-30-10#	Upper Compt Purge Air Supply	≤ 4
FCV-30-14#	Lower Compt Purge Air Supply	≤ 4
FCV-30-15#	Lower Compt Purge Air Supply	≤ 4
FCV-30-16#	Lower Compt Purge Air Supply	≤ 4
FCV-30-17#	Lower Compt Purge Air Supply	≤ 4

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TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Phase "A" Containment Vent Isolation (Cont.)		
FCV-30-19 #	Inst Room Purge Air Supply	≤ 4
FCV-30-20 #	Inst Room Purge Air Supply	≤ 4
FCV-30-37 #	Lower Compt Pressure Relief	≤ 4
FCV-30-40 #	Lower Compt Pressure Relief	≤ 4
FCV-30-50 #	Upper Compt Purge Air Exh	≤ 4
FCV-30-51 #	Upper Compt Purge Air Exh	≤ 4
FCV-30-52 #	Upper Compt Purge Air Exh	≤ 4
FCV-30-53 #	Upper Compt Purge Air Exh	≤ 4
FCV-30-56 #	Lower Compt Purge Air Exh	≤ 4
FCV-30-57 #	Lower Compt Purge Air Exh	≤ 4
FCV-30-58 #	Inst Room Purge Air Exh	≤ 4
FCV-30-59 #	Inst Room Purge Air Exh	≤ 4
FCV-90-107	Cntmt Bldg LWR Compt Air Mon	≤ 5
FCV-90-108	Cntmt Bldg LWR Compt Air Mon	≤ 5
FCV-90-109	Cntmt Bldg LWR Compt Air Mon	≤ 5
FCV-90-110	Cntmt Bldg LWR Compt Air Mon	≤ 5
FCV-90-111	Cntmt Bldg LWR Compt Air Mon	≤ 5
FCV-90-113 #	Cntmt Bldg Up Compt Air Mon	≤ 5
FCV-90-114 #	Cntmt Bldg Up Compt Air Mon	≤ 5
FCV-90-115 #	Cntmt Bldg Up Compt Air Mon	≤ 5
FCV-90-116 #	Cntmt Bldg Up Compt Air Mon	≤ 5
FCV-90-117 #	Cntmt Bldg Up Compt Air Mon	≤ 5

* Not subject to Type C leakage tests.

Provisions of LCO 3.0.4 are not applicable if the valve is secured in its isolated position with power removed.

Provisions of LCO 3.0.4 are not applicable if the valve is secured in its isolated position with power removed and either FCV-62-73 or FCV-62-74 is maintained operable.

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3/4 6-30 and 3/4 6-31

TVA's submittal of June 19, 1984 included a proposal to revise the ice condenser inlet door surveillance frequency (SR 4.6.5.3.1) to a 9 month interval to coincide with the required ice-weighing outage. This proposal was discussed with NRC Staff representatives at a TVA/NRC Technical Specification meeting of August 7-8, 1984.

NRC Staff represenatives indicated that to offset the increase in the test interval, the numbers of doors to be tested would need to be increased from 25 to 50 percent. TVA indicated that the increase in precentage of test population was acceptable. Specification 4.6.5.3.1 should be revised accordingly.

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LIMITING CONDITION FOR OPERATION

3.6.5.3 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be closed and OPERABLE.

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

ACTION:

With one or more ice condenser doors open or otherwise inoperable, POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained less than or equal to 27°F; otherwise, restore the doors to their closed positions or OPERABLE status (as applicable) within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

a. Continuously monitored and determined closed by the inlet door position monitoring system, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is initially fully-loaded and at least once per ~~6~~ 9 months thereafter by:

1) Verifying that the torque required to initially open each door is less than or equal to 675 inch pounds;

2) Verifying that opening of each door is not impaired by ice, frost or debris;

3) Testing a sample of at least ~~25%~~ 50% of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional torque component. The doors selected for determination of the "door opening torque" shall be selected to ensure that all doors are tested at least once during ~~four~~ two test intervals;

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SURVEILLANCE REQUIREMENTS (Continued)

4) Testing a sample of at least ~~25%~~ 50% of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as ~~the~~ "door closing torque" and is equal to the nominal door torque minus a frictional torque component. The doors selected for determination of the "door closing torque" shall be selected to ensure that all doors are tested at least once during ~~four~~ two test intervals; and

5) Calculation of the frictional torque of each door tested in accordance with Specifications 4.6.5.3.1b.3) and 4) above. The calculated frictional torque shall be less than or equal to 40 inch-pounds.

4.6.5.3.2 Intermediate Deck Doors - Each ice condenser intermediate deck door shall be:

a. Verified closed and free of frost accumulation by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is initially fully-loaded and at least once per 18 months thereafter by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

	Door	Lifting Force
1)	0-1, 0-5	≤ 37.4 lbs.
2)	0-2, 0-6	≤ 33.8 lbs.
3)	0-3, 0-7	≤ 31.8 lbs.
4)	0-4, 0-8	≤ 31.0 lbs.

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 92 days by visually verifying:

a. That the doors are in place, and

b. That no condensation, frost, or ice has formed on the doors or blankets which would restrict their lifting and opening if required.

Specification 4.7.10.3 requires the submittal of an annual report if removable contamination from a sealed source or fission detector is greater than or equal to .005 microcurie. This reporting requirement is contrary to NUREG-1024, Section 3.10 which states,

> "Some of the requirements in the Standard Technical Specifications (STS) are for the purpose of collecting information and do not add to operational safety. The Technical Specifications should not contain such requirements."

The scope of this annual report is not specified which leads to the conclusion that Specification 4.7.10.3 is included as a punitive requirement which is also cited in NUREG-1024, Section 3.2 as an STS problem area,

> "The ACTION Statements of some technical specifications seem to be structured as a punitive measure against utilities that have safety equipment out of service rather than as a function of the significance of the equipment outage from the standpoint of risk to the public."

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SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

~~4.7.10.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.~~

3/4 7-29, 3/4 7-30, 3/4 7-31, and 3/4 7-32

Surveillance Requirement 4.7.11.1.e.4 should be revised to specify a fire suppression pump starting pressure of 105 psig.

Surveillance Requirements 4.7.11.1 and 4.7.11.2 should be revised as indicated. The requirements to cycle valves every 12 months should be modified to require testing of only those valves that are non-self indicating. This proposed change was discussed with NRC Staff representatives during a TVA/NRC Technical Specification meeting of August 7-8, 1984 in which it was established that the intent of the testing was to establish and/or verify valve position.

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3/4.7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11.1 The Fire Suppression Water System shall be OPERABLE with:

a. At least two fire suppression pumps, each with a capacity of 1590 gpm at 330 feet of head, with their discharge aligned to the fire suppression header, and

b. An OPERABLE flow path capable of taking suction from the forebay and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the standpipe hose valves, and the first valve upstream of the water flow device on each Spray System required to be OPERABLE per Specifications 3.7.11.2 and 3.7.11.4.

APPLICABILITY: At all times.

ACTION:

a. With only one pump OPERABLE, restore at least two pumps to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable establish a backup Fire Suppression Water System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.11.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by starting each pump and operating it for at least 15 minutes on recirculation flow,

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

c. At least once per 6 months by performance of a system flush,

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

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SURVEILLANCE REQUIREMENTS

e. 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 automatic valve in the flow path actuates to its correct position,

2) Verifying that each pump develops at least 1590 gpm at a system head of 330 feet,

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

4) Verifying that each fire suppression pump starts as designed to maintain the Fire Suppression Water System pressure at the pump discharge greater than or equal to ~~125~~ 105 psig.

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

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SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. Reactor building - RC pump area, Annulus;

b. Auxiliary building - Elev. 692, 713, 729, 737, 757, 772, 782, ABGTS Filters, EGTS Filters, Purge Filters, 125 V Battery Rooms;

c. Control building - Elev. 692, Cable spreading room, MCR air filters and Operator living area;

d. Diesel building - Corridor area;

e. Turbine building - Control building wall; and

f. ERCW pumping station (Intake).

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.11.2 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 non-self indicating valve (manual, power-operated, or automatic) in the flow path is in its correct position,

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

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SURVEILLANCE REQUIREMENTS (Continued)

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 automatic valves in the flow path actuate to their correct positions on a single or cross zoned test signal as designed, and

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

2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3) By a visual inspection of each sprinkler head/spray nozzle's spray area to verify the spray pattern is not obstructed.

3/4 8-1, 3/4 8-2, 3/4 8-3, 3/4 8-4, 3/4 8-6, 3/4 8-7, 3/4 8-8, 3/4 8-9, 3/4 8-10, 3/4 8-11 and 3/4 8-12

Specifications 3.8.1.1 and 4.8.1.1.2 - For clarity revise where indicated to designate the diesel generator(s) as set(s).

Specification 3.8.1.1 ACTION Statements a., b., and d. - Revise as indicated in accordance with NRC Generic Letter 84-15.

Specification 3.8.1.1 ACTION Statement f. - Incorporate the proposed ACTION Statement to defer diesel starting when one or more diesels are inoperable solely because the fuel levels are below minimum. Twenty-four hours has been proposed to restore fuel levels before additional surveillance is required. TVA believes this approach is consistent with the philosophy expressed in NRC Generic Letter 84-15.

Specification 4.8.1.1.2 - Incorporate the proposed footnote in accordance with NRC Generic Letter 84-15. This change limits the number of cold fast starts.

Surveillance Requirement 4.8.1.1.2.f.10 - This SR should be deleted since Watts Bar does not have installed cross connection lines.

Surveillance Requirement 4.8.1.1 - Replace with the diesel generator reliability improvement program in accordance with NRC Generic Letter 84-15.

Surveillance Requirements 4.8.1.1.2.e and f - For bases see TVA submittal of June 19, 1984.

Surveillance Requirement 4.8.1.1.2.d.11.c - Incorporate the emergency stop feature.

Surveillance Requirements 4.8.1.1.3.a.3 and 4.8.1.1.3.b - TVA has reviewed the battery Surveillance Requirements associated with the diesel generator batteries. The batteries consist of 57 cells. The minimum float is 2.2 volts per cell or 125.4 volts across the bank. The float voltage range is 2.20 volts to 2.25 volts. The minimum voltage is 1.75 volts per cell and the maximum voltage is 2.39 volts per cell. These result in a minimum and maximum voltage across the bank of 99.75 volts and 136.23 volts, respectively. The Final Safety Analysis Report specifies requirements for the battery chargers. These must meet or exceed the battery requirements specified in the Technical Specifications.

Surveillance Requirements 4.8.2.1.a.2 and 4.8.2.1.b - TVA has reviewed the battery Surveillence Requirements associated with the diesel generator batteries. The batteries consists of 60 cells. The critical voltage parameters are the same as those listed for the diesel generator batteries. The float voltage and overvoltage limits should be revised as indicated. TVA has reviewed the battery charger capacity. The value specified in Surveillance Requirement 4.8.2.1.c.4 is consistent with the expected load on the charger during normal operation.

Surveillance Requirement 4.8.1.1.2.a.4.c - Delete the requirement to start the diesel by a loss of offsite power signal coincident with a safety injection signal. The actual logic circuit start circuit consists of three sets of contacts wired in series. One set is the safety injection signal and two sets

are the loss of offiste power signal. Opening any one of the contacts will result in a diesel start. The coincident test requires that at least two of the contacts be opened at the same time. The test really is unnecessary and repetitive. The wiring diagram for this position of the circuit is shown in FSAR Figure 8.3-29. Relay K609 is the safety injection signal and relays 27D1AX and 27D1AY are the loss of offsite power signal.

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution ~~system, and~~

b. Four separate and independent diesel generator~~s~~ sets, each with:

2 ~~1~~) Two separate engine-mounted fuel tanks ~~containing~~ a minimum volume of 250 gallons of fuel in each tank,

3 ~~2~~) A separate 7 day fuel storage tank containing a minimum volume of 62,000 gallons of fuel,

4 ~~3~~) A separate fuel transfer pump, and

5 ~~4~~) A separate 125-volt DC distribution panel, 125-volt D.C. battery bank and associated charger.

1) Two diesel engines driving a common generator set

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

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, ~~demonstrate~~ the OPERABILITY of the remaining A.C. offsite sources by performing Specification~~s~~ 4.8.1.1.1a. ~~and 4.8.1.1.2a.4)~~ within 1 hour and at least once per 8 ~~hours thereafter;~~ restore at least two offsite circuits and four diesel generator~~s~~ sets 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.

and Specification 4.8.1.1.2.a.4 within 24 hours;

b. With one offsite circuit and one diesel generator set of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite sources by performing Specification~~s~~ 4.8.1.1.1a. ~~and 4.8.1.1.2a.4)~~ within 1 hour and at least once per 8 hours thereafter; ~~restore at least~~ one of the inoperable ~~sources to OPERABLE status~~ within 12 hours or be in at least ~~HOT STANDBY~~ within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ~~Restore~~ at least two offsite circuits and four diesel generator~~s~~ sets to OPERABLE status within 72 hours from the time of initial loss ~~or~~ be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

and Specification 4.8.1.1.2.a.4 within 8 hours;

c. With one diesel generator set inoperable in addition to the requirements of ACTION a. or b. above, verify that:

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator set as a source of emergency power are also OPERABLE, and

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of four diesel generators by performing Specification 4.8.1.1.2a.4) within 1 hours and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With either Diesel Generator 1A-A or 2A-A inoperable simultaneously with 1B-B and/or 2B-B of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Specification 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable 1A-A and 2A-A or 1B-B and 2B-B to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least four diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Insert ACTION statement f

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the engine-mounted fuel tank,

Proposed Action Statement F

With one or more diesel generators inoperable solely because the fuel levels in one or more tanks are below the minimum, restore the levels to above the minimum within 24 hours; otherwise comply with ACTION Statements a, b, or e above as applicable.

* The diesel generator set start (10 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of surveillance testing may be preceeded by an engine prelube period and/or other warmup procedures recommended by the manufactur so that mechanical stress and wear on the diesel engine is minimized.

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SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying the fuel level in the 7-day fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the 7-day fuel storage tank to the engine-mounted tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least ~~900~~ 900 ± 18 rpm in less than or equal to 10 seconds.* The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator sets shall be started for this test by using one of the following signals:

   a) Manual, or

   ~~b) Simulated loss-of-offsite power by itself, or~~

   ~~c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal, or~~

   c ~~d)~~ An ESF actuation test signal ~~by itself~~.

5) Verifying the generator is synchronized, loaded to greater than or equal to 4400 kW in less than or equal to 60 seconds,* and operates with a load greater than or equal to 4400 kW for at least 60 minutes, and

6) Verifying the diesel generator set is aligned to provide standby power to the associated shutdown boards.

b. ~~1)~~ At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the engine-mounted fuel tanks;

c. ~~2)~~ At least once per 31 days by checking for and removing accumulated water from the 7 day fuel oil storage tanks;

~~c. At least once per 92 days and from new fuel oil prior to its addition to the 7-day storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:~~

~~1) A water and sediment content of less than or equal to 0.05 volume percent;~~

~~2) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.3 centistokes;~~

~~3) A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity of 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees;~~

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SURVEILLANCE REQUIREMENTS (Continued)

~~4) An impurity level of less than 2 mg. of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70, analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil, and~~

~~5) The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977, analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.~~

d. At least once per 18 months during shutdown by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

2) Verifying the generator capability to reject a load of greater than or equal to 600 kW while maintaining voltage at 6900 ± 690 volts and frequency at 60 ± 1.2 Hz;

3) Verifying the generator capability to reject a load of 4400 kW without tripping. The generator voltage shall not exceed 7866 volts during and following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the shutdown boards and load shedding from the shutdown boards, and

b) Verifying the diesel starts on the auto-start signal, energizes the shutdown boards with permanently connected loads within 10 seconds, energizes the auto-connected

Insert SR 4.8.1.1.2.e. and f.

Surveillance Requirement 4.8.1.1.2

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM D2276-78, and verifying that particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78.

f. By sampling new fuel oil in accordance with ASTM D4057 prior to addition to storage tanks and:

(1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the 7 day storage tanks that the sample has:

(a) An API Gravity of within 0.3 degress at 60 $^{\circ}$ F or a specific gravity of within 0.0016 at 60/60 $^{\circ}$ F, when compared to the supplier's certiciate or an absolute specific gravity of 60/60 $^{\circ}$ F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees.

(b) A kinematic viscosity at 40 $^{\circ}$ C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravtiy was not determined by comparison with the supplier's certification.

(c) A flash point equla to or greater than 125{o}F, and

(d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

(2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

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SURVEILLANCE REQUIREMENTS (Continued)

these limits during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2d.6)b);*

8) Verifying that the auto-connected loads to each diesel generator set do not exceed the 2000-hour rating of 4840 kW;

9) Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

~~10) Verifying that the fuel transfer pump transfers fuel from each 7-day fuel storage tank to the engine-mounted tanks of each diesel via the installed cross-connection lines;~~

10 ~~11)~~ Verifying that the automatic load sequence timers are OPERABLE and their setpoints are within the specified bands;

11 ~~12)~~ Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

a) Engine overspeed, or

b) 86 GA lockout relay, or

c) Emergency stop

9. ~~e.~~ At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously, during shutdown, and verifying that all diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds; and

*If Specification 4.8.1.1.2d.6)b) is not satisfactorily completed, it is not necessary to ~~report~~ repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 4400 kW for 1 hour or until operating temperature has stablized.

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SURVEILLANCE REQUIREMENTS (Continued)

h. ~~f.~~ At least once per 10 years by:

1) Draining each 7-day fuel storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 The 125-volt D.C. distribution panel, 125-volt D.C. battery bank and associated charger for each diesel generator shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying:

1) Correct breaker alignment, indicated power availability and voltage on the distribution panels greater than or equal to 118 volts,

2) That each battery bank and charger meet the Category A limits in Table 4.8-2 of Specification 4.8.2.1, and

3) That the total battery terminal voltage is greater than or equal to ~~128~~ 126 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with a battery terminal voltage below ~~104~~ 100 volts or a battery overcharge with a battery terminal voltage above ~~143~~ 136 volts by:

1) Verifying that the parameters in Table 4.8-2 of Specification 4.8.2.1 meet the Category B limits,

2) Verifying there is no visible corrosion at either terminals or connectors, or the cell to terminal connection resistance of these items is less than 150 x 10^{-6} ohms, and

3) Verifying that the average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The battery to battery and terminal connections are clean, tight and coated with anti-corrosion material, and

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SURVEILLANCE REQUIREMENTS (Continued)

3) The resistance of each cell to terminal connection is less than or equal to 150 x 10^{-6} ohms.

~~4.8.1.1.4 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Report of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.~~

→ Insert SR 4.8.1.1.4

4.8.1.1.4 Diesel Generator Reliability Improvement Program

As a minimum the Reliability Improvement Program report for NRC audit shall include:

(a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

(b) analysis of failures and determination of root causes of failures

(c) evaluation of each of the recommendations of NUREG/CR-0660, Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors, with respect to their application to the Plant

(d) identification of all actions taken or to be taken to (1) correct the root causes of failures defined in (b) above and (2) achieve a general improvement of diesel generator reliability

(e) the schedule for implementation of each action from (d) above

(f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

Once a licensee has prepared and maintained an initial report detailing the diesel generator reliability improvement program at his site, as defined above, the licensee need prepare only a supplemental report within 30 days after each failure during a valid demand for so long as the affected diesel generator unit contiues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

In addition to the above, submit a yearly data report on the diesel generator reliability.

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TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST ~~100~~ 20 VALID TESTS*	TEST FREQUENCY
≤ 1	At least once per 31 days
≥ 2	At least once per ~~14~~ 7 days**
~~3~~	~~At least once per 7 days~~
~~≥ 4~~	~~At least once per 3 days~~

diesel generator set

number of tests and failures are

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the ~~last 100 tests are~~ determined on a per ~~nuclear unit~~ basis. For the purposes of this test schedule, only valid tests conducted after ~~the OL~~ 20 issuance date shall be included in the computation of the "last ~~100~~ 20 valid tests." Entry into this test schedule shall be made at the 31-day test frequency.

** This test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one or less.

TABLE 4.8-2

ADDITIONAL RELIABILITY ACTIONS

No. of failures in last 20 valid test	No of failures in last 100 valid tests	Action
3	6	Within ~~14~~ 30 days prepare and maintain a report for NRC audit, describing the diesel generator reliability ~~improvement~~ program implemented at the site, in accordance with Surveillance Requirement 4.8.1.1.4. ~~Minimum requirements for the report are indicated in Attachment 1 to this table.~~
5	11	Declare the diesel generator inoperable and perform ~~Perform~~ a requalification test program for the affected diesel generator, pursuant to the Attachment to this table. ~~Requalification test program requirements are indicated in Attachment 2 to this table.~~

ATTACHMENT TO TABLE 4.8-2
DIESEL GENERATOR REQUALIFICATION PROGRAM

(1) Perform seven consecutive successful demands without a failure within 30 days of diesel generator being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of diesel generator of being restored to operable status.

(2) If a failure occurs during the first seven tests in the requalification test program, perform seven successful demands without an additional failure within 30 days of diesel generator of being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of being restored to operable status.

(3) If a failure occurs during the second seven tests (tests 8 through 14) of (1) above, perform fourteen consecutive successful demands without an additional failure within 75 days of the failure which occurred during the requalification testing.

(4) Following the second failure during the requalification test program, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

(5) During requalification testing the diesel generator should not be tested more frequently than at 24-hour intervals.

After a diesel generator has been successfully requalified, subsequent repeated requalification tests will not be required for that diesel generator under the following conditions:

(a) The number of failures in the last 20 valid demands is less than 5.

(b) The number of failures in the last 100 valid demands is less than 11.

(c) In the event that following successful requalification of a diesel generator, the number of failures is still in excess of the remedial action criteria (a and/or b above) the following exceptior will be allowed until the diesel generator is no longer in violation of the remedial action criteria (a and/or b above).

Requalification testing will not be required provided that after each valid demand the number of failures in the last 20 and/or 100 valid demands has not increased. Once the diesel generator is no longer in violation of the remedial action criteria above the provisions of those criteria alone will prevail.

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LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. ~~Diesel~~ Generator Sets 1A-A and 2A-A or 1B-B and 2B-B each with:
1) Two diesels driving a common generator,

2 ~~1)~~ Two Engine-mounted fuel tanks containing a minimum volume of 250 gallons of fuel ~~per tank~~,

3 ~~2)~~ A 7 day fuel storage tank containing a minimum volume of 62,000 gallons of fuel,

4 ~~3)~~ A fuel transfer pump, and

5 ~~4)~~ A separate 125-volt DC distribution panel, 125-volt DC battery bank, and associated charger.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 3 square inch vent. In addition, when in MODE 5 with the Reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements ~~of~~ Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.~~5~~ 5), 4.8.1.1.3, and 4.8.1.1.4.

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3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt Battery Bank No. I, and its associated full capacity charger,

b. 125-Volt Battery Bank No. II, and its associated full capacity charger,

c. 125-Volt Battery Bank No. III, and its associated full capacity charger, and

d. 125-Volt Battery Bank No. IV, and its associated full capacity charger.

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

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Specification 4.8.2.1a.1) within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to ~~128~~ 132 volts on float charge.

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b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts, or battery overcharge with battery terminal voltage above ~~140~~ 143 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3) The average electrolyte temperature of 12 connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

4) The battery charger will supply at least 150 amperes at 125 volts for at least 4 hours.

d. At least once per 18 months by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

3/4 9-3

Specification 3.9.2 should be revised as indicated to require that the source range monitors be operating in mode 6.

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3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Soruce Range Neutron Flux Monitors shall be OPERABLE and operating each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4 9-6

Specifications 3.9.5 and 4.9.5 should be revised as recommended by NRC OIE for clarification of communication requirements.

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3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct continuous communications shall be maintained open between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct continuous communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

open

3/4 9-7

Specification 4.9.6.1 should be revised as indicated for clarification of the refueling machine teting requirements. The mechanical load cutoff valve is not repeatable. The proposed wording will allow TVA to set the interlock for repeatability.

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3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine used for movement of fuel assemblies having:

1) A minimum capacity of 3150 pounds,

2) An electrical overload cutoff limit less than or equal to 2850 pounds, and

3) A mechanical overload cutoff limit less than or equal to 3400 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

1) A minimum capacity of 700 pounds, and

2) A load indicator which shall be used to prevent lifting loads in excess of 690 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for refueling machine and/or hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3150 pounds and demonstrating an automatic electrical load cutoff when ~~the~~ crane load exceeds 2850 pounds and an automatic mechanical load cutoff ~~when~~ the crane load exceeds 3400 pounds.

before

4.9.6.2 Each auxiliary hoist and associated load indicator ~~used for~~ movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.

Specification 3.9.8.1 should be revised as indicated. During RHR system operation, the RHR pumps take suction from loop 4 hot leg. Discharge back to the vessel is through the cold legs on loops 1 and 4. The safety injection pump cold leg discharge lines tie into the RHR cold leg lines in between two check valves. With RHR running in normal mode the pressure at the point of safety injection tie in will be higher in loops 1 and 4 than in loops 2 and 3. It is not possible to balance the Safety Injection System in this condition. The test takes approximately one half hour to perform. Both loops of RHR would be shut down for this period of time.

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3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration ~~of the~~ Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible.** Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

** During ECCS flow balance testing of the Safety Injection System both RHR loops may be removed from operation provided the other requirements of the ACTION statement are met.

3/4 9-11

Please refer to TVA's submittal of June 19, 1984 which included a request to delete the reference to manual initiation of containment vent isolation from SR 4.9.9. It was inadvertently stated that Watts Bar did not have manual initiation of containment vent isolation. The correct basis for the requested deletion is that the manual initiation of the Containment Ventilation Isolation System is accomplished by means of the manual phase A isolation switch in the Control Room. The initiation of a phase A isolation every 7 days during refueling is not practical. Phase A isolates most of the process lines entering containment. Isolation of these lines would be disruptive to the ongoing work.

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3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

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

ACTION:

With the Containment Ventilation Isolation System inoperable, close each of the ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE ~~within 100 hours prior~~ to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ventilation isolation occurs ~~on manual initiation and~~ on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

3/4 11-13

TVA's submittal of June 19, 1984 included a request to revise table notation 3 of Table 4.11-2 to require sampling based on noble gas monitor readings should activity increase by a factor of 3.

Several conference calls have recently been held between TVA and NRC representatives to discuss this proposal. As a result of these discussions, TVA now proposes that this requirement be more clearly defined as indicated.

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7, in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3) Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(5) Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(6) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

(7) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

(8) During releases via this Exhaust System.

(9) In MODES 1, 2, 3, and 4, the upper and lower compartments of the containment shall be sampled prior to PURGING. Prior to breaking CONTAINMENT INTEGRITY in MODES 5 and 6, the upper and lower compartments of the containment shall be sampled. The incore instrument room purge sample shall be obtained at the shield building exhaust between 5 and 10 minutes following initiation of the incore instrument room purge.

(10) Prior to VENTING in MODES 1, 2, 3, and 4, the upper and lower compartments of the containment shall be sampled daily when VENTING is to occur on that day.

(11) Not applicable to the Shield Building Exhaust.

(12) Not applicable when the most recent Secondary Coolant System specific activity sample and analysis program gross radioactivity determination is less than 1×10^{-6} μCi/gm provided the discharge Radiation Monitor Setpoint is less than or equal to 1×10^{-6} μCi/ml above background.

B 3/4 1-3

Bases 3/4.1.3 should be revised as indicated to clarify the fully withdrawn position for the shutdown banks. Fully withdrawn is defined as 225 steps rather than the 228 steps that are used for the control banks. This position will minimize the potential for vibratory interactions between the rodlets and control rod guide tubes. The shutdown banks worths will be unaffected because the differential rod worths are very low in this region.

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BORATION SYSTEMS (Continued)

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank (RWST).

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The shutdown banks will have the fully withdrawn position defined as 225 steps rather than 228 steps. This will minimize the potential for vibratory interaction between the rodlets and the control rod guide tubes.

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required in MODES 1 and 2 to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod in MODES 1 and 2 requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

B 3/4 4-1

TVA's submittal of June 19, 1984 included a request to modify Specification 3.4.1.2 based upon the Westinghouse notification of an inconsistency in the Technical Specifications and the mode 3 safety analyses for steamline break, control rod ejection, and bank withdrawal from a subcritical condition.

Bases 3/4.4.1 should also be revised accordingly.

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 310°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The bank withdrawal from a subcritical condition accident requires two reactor coolant pumps be running. With only one reactor coolant pump running, the rod control system must be made incapable of bank withdrawal.

B 3/4 6-3

Bases 3/4.6.1.9 should be revised as indicated for the purpose of clarification.

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SHIELD BUILDING STRUCTURAL INTEGRITY (Continued)

for the life of the facility. Structural integrity is required to provide: (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

3/4.6.1.8 EMERGENCY GAS TREATMENT SYSTEM

The OPERABILITY of the EGTS ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. This requirement is necessary to meet the assumptions used in the accident analyses and limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during LOCA conditions. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM

The ~~use of the containment purge lines is restricted to the 24-inch at 70% open and the 12-inch~~ purge supply and exhaust isolation valves ~~since the 24-inch valves at 70% open and the 12-inch valves~~ are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation with one line open will be limited to 500 hours during a calendar year. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activi-ies, may be used to support the additional time requests.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of the valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

Furthermore, the restriction on opening of the 24-inch purge valves further ensures valve closure during these accidents.

B 3/4 6-5

Bases 3/4.6.5.3 should be revised as indicated to provide clarification on the number of ice condenser doors that can be blocked for maintenance.

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3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 15 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will: (1) be distributed evenly through the containment bays, (2) contain sufficient boron to preclude dilution of the containment sump following the LOCA, and (3) contain sufficient heat removal capability to condense the Reactor Coolant System volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1399 pounds of ice per basket contains a 10% conservative allowance for ice loss through sublimation which is a factor of 10 higher than assumed for the ice condenser design and 1% for weighing accuracies. In the event that observed sublimation rates are equal to or lower than design predictions after three years of operation, the minimum ice baskets weight may be adjusted downward. In addition, the number of ice baskets required to be weighted each 9 months may be reduced after 3 years of operation if such a reduction is supported by observed sublimation data.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the Ice Bed Temperature Monitoring System ensures that the capability is available for monitoring the ice bed temperature. The ice bed temperature may be determined at the local ice condenser temperature monitoring panel as well as in the main Control Room and the Monitoring System would still be considered OPERABLE. In the event the Monitoring System is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.3 ICE CONDENSER DOORS

The OPERABILITY of the ice condenser doors and the requirement that they be maintained closed ensures that the Reactor Coolant System fluid released during a LOCA will be diverted through the ice condenser bays for heat removal and that excessive sublimation of the ice bed will not occur because of warm air intrusion. The number of lower inlet, intermediate deck, and/or top deck doors that may be blocked for maintenance will be evaluated by plant personnel for its effect on containment accident response.

5-2 and 5-3

Figures 5.1-1 and 5.1-2 have been revised to improve clarity.

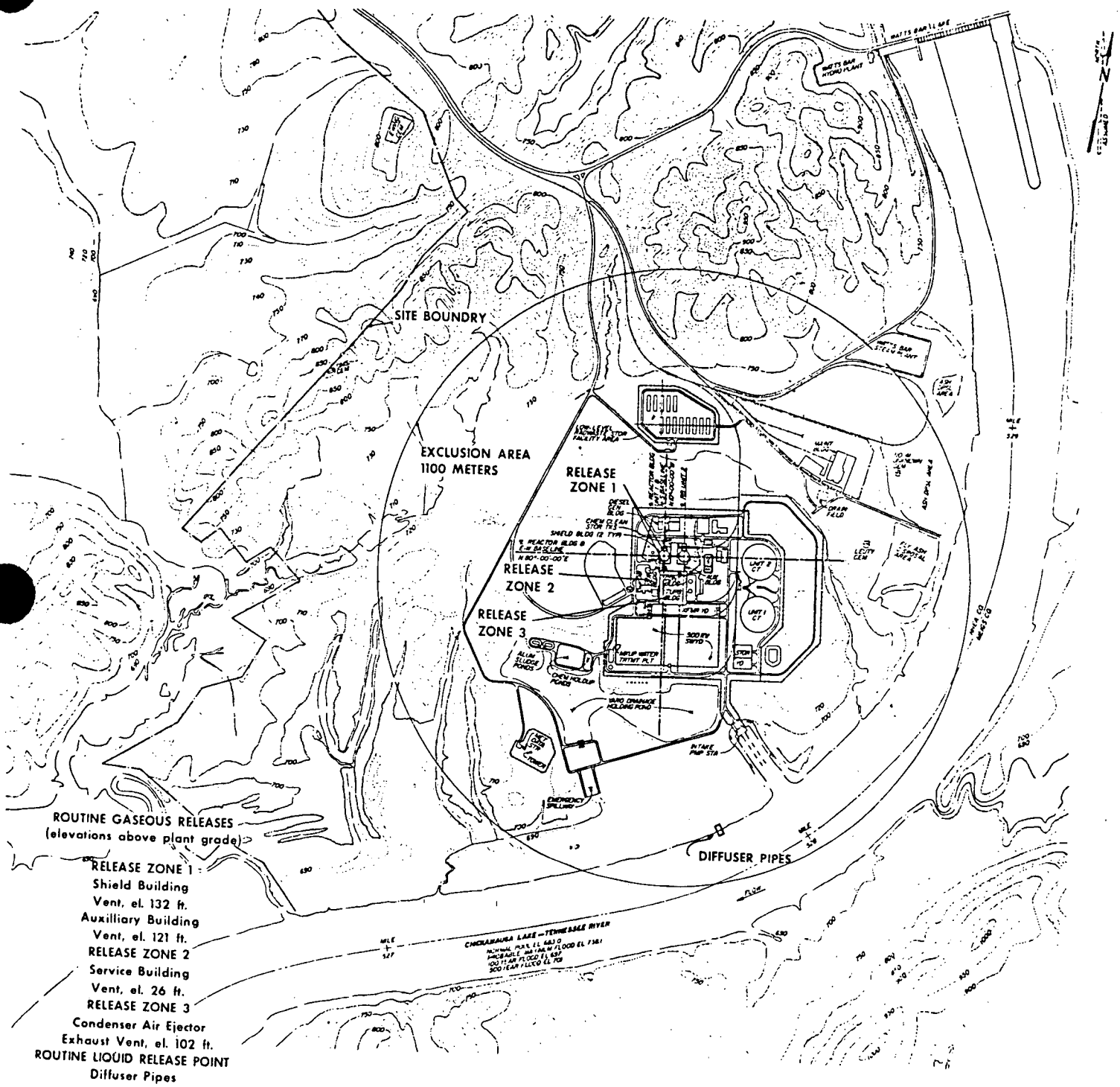

FIGURE 5.1-1

SITE BOUNDARY, EXCLUSION AREA,
AND RADIOACTIVE EFFLUENT RELEASE POINTS

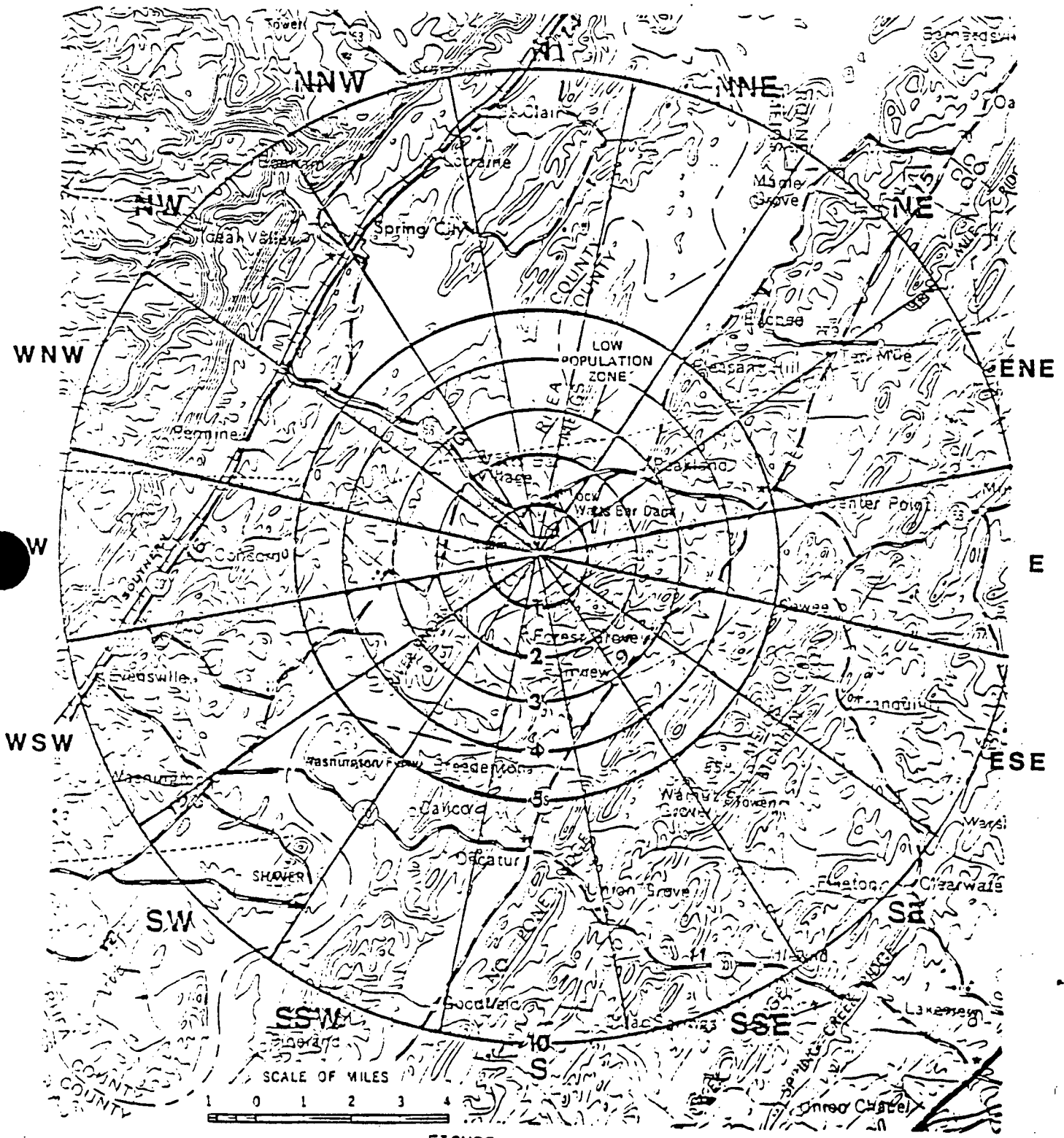

FIGURE 5.1-2

LOW POPULATION ZONE

6-8 and 6-9

TVA's submittal of June 19, 1984 included proposed modifications to Section 6 πAdministrative Controlsπ to reflect the recent TVA reorganization. As part of the Section 6 proposal, TVA requested that the PORC composition and quorum be revised.

Please note that the attached marked-up pages are from a post proof and review version of the Technical Specifications.

TVA now proposes that a variable membership for the PORC be adopted. The PORC will be composed of a minimum of five members but in no case more than ten. The quorum requirements require revising to address the variable membership and clarify the fact that the quorum can be composed from the total membership and not just the minimum membership.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Operations and Engineering Superintendent and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ISEG.

6.5 REVIEW AND AUDIT

6.5.0 The Manager, Office of Nuclear Power, is responsible for the safe operation of all TVA nuclear power plants. The functional organization for review and cognizance of audits is shown on Figure 6.2-1.

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of at least ~~the following members, not to exceed ten members:~~ at least a minimum of the following but in no case more than ten members:

Chairman:	Plant Manager or his Alternate
Member:	Operations Supervisor
Member:	Engineering Supervisor
Member:	Maintenance Supervisor (Electrical, Mechanical, or Instrumentation)
Member:	Health Physics Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

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QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and ~~at least half of the total members including alternates.~~ three members of the total membership or one more than half of the total membership, including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of: (1) all procedures required by Specification 6.8.1 and changes thereto, (2) all programs required by Specification 6.8.5, and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix "A" Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Site Director, and to the Chief, Nuclear Safety Staff;

f. Review of all REPORTABLE EVENTS;

g. Review of unit operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the Chief, Nuclear Safety Staff;

i. Review of the Plant Physical Security Plan and implementing procedures and shall submit recommended changes to the Chief, Nuclear Safety Staff;

j. Review of the Site Radiological Emergency Plan and implementing procedures and shall submit recommended changes to the Chief, Nuclear Safety Staff;