TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

April 9, 1985

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 No. 50-390
Tennessee Valley Authority	)	

By letter dated December 11, 1984, NRC forwarded to TVA the Final Draft Technical Specifications for the Watts Bar Nuclear Plant Unit 1. By letter dated February 15, 1985 NRC forwarded various revised pages to the Final Draft Technical Specifications. In the February 15, 1985 letter NRC also requested that TVA certify under oath and affirmation that the December 11, 1984 technical specifications, as modified by the February 15, 1985 letter changes (excluding section 6), accurately reflect the as-built facility, the FSAR (as amended), and the SER analysis.

Accordingly, to the best of my knowledge, I certify that the unit 1 technical specifications, excluding section 6, as described above, accurately reflect the as-built facility as described in the FSAR as amended, and the SER analysis except where indicated in enclosures 1 and 2. (Section 6 will be reviewed for certification upon receipt from NRC.)

Enclosure 1 contains technical specification changes which must be resolved for certification of the unit 1 technical specifications. Enclosure 2 contains a list of SER sections which require updating in order for the unit 1 technical specifications to be certified to the SER. Enclosure 3 is a list of technical specification changes which are considered a further enhancement and optimization by TVA. These changes are not required for certification, but would improve operator use and understanding, provide for more efficient operation, and eliminate unnecessary requirements.

If you have any questions, please get in touch with me at FTS 858-2688.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. H. Shell

R. H. Shell
Nuclear Engineer

Sworn to and subscribed before me
this 9th day of April 1985

Paulette H. White

Notary Public
My Commission Expires 8-24-88

Enclosure
cc: See page 2

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Director of Nuclear Reactor Regulation April 9, 1985

cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attn: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT

TECHNICAL SPECIFICATION CHANGES
REQUIRED FOR CERTIFICAITON

Reactor Coolant System - Hot Shutdown

References: 1) Letter from E. G. Adensam to H. G. Parris dated August 22, 1984

2) Letter from J. A. Domer to E. Adensam dated January 3, 1985

In reference 1 NRC questioned the adequacy of the FSAR analysis for an uncontrolled rod withdrawal event in mode 4 with one residual heat removal pump in operation (question 28). TVA responded that this issue was being addressed generically by the industry/NRC effort to improve and optimize technical specifications in reference 2. TVA believed that this method of resolution was acceptable to NRC. However, TVA now understands from recent discussions with the reviewers that NRC's position is that this issue must be addressed for fuel load to avoid any material false statements. Consequently, TVA is providing a marked-up technical specification page to require that two reactor coolant pumps be in operation in mode 4 whenever the reactor trip breakers are closed. If TVA's understanding of this problem is in error, TVA prefers to delay implementation of this change until generic efforts to improve and optimize technical specifications are concluded. TVA does not believe that an uncontrolled rod withdrawal event is credible in mode 4 because the rod control system is in manual mode and not subject to spurious operation.

REACTOR COOLANT SYSTEM

FINAL DRAFT

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation. Two reactor coolant pumps shall be in operation when the Reactor Trip System Breakers are closed.**

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*

e. RHR Loop A, and

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

c. With less than two reactor coolant pumps in operation and the Reactor Trip System Breakers in the closed position, within 1 hour open the Reactor Trip System Breakers.

*A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 310°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

**All Reactor Coolant pumps and residual heat removal pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

Source Range Response Time

References: 1) Letter from E. G. Adensam to H. G. Parris dated August 22, 1984

2) Letter from J. A. Domer to E. Adensam dated January 3, 1985

In reference 1 NRC requested additional information regarding rod withdrawal accidents in the shutdown condition (question 27). TVA responded in reference 2 that both source range detectors are required to be operable in the shutdown condition and that they provide sufficient protection with sufficient margin. TVA understands from discussions with the reviewers that it is NRC's position that Table 3.3-2 of the technical specifications should be revised to include a response time for the source range channel.

TVA has reviewed the matter and believes inclusion of a source range response time is not warranted to improve safety. This is based on the fact that the only additional portion of the system tested by this requirement would be the rack electronic equipment for the source range channel. The neutron detectors are exempt from testing, and the reactor trip breakers are timed once and added to the individual channel response times to determine overall response times. The industry experience has been that the response time of the electronic equipment has been stable and does not change.

However, if NRC believes it is important to list a source range channel response time, TVA will comply with NRC requirements. A source range response time has not been explicitly modeled in the rod withdrawal accident. Only the bounding case analysis using the power range low setpoint has been evaluated. TVA has estimated the response time for the source range channel from FSAR figure 15.2-1. The source range setpoint is approximately six decades lower than the power range low setpoint. The source range channel has this additional time to respond over the power range channel based on the technical specification definition of response time. The source range channel exceeds the setpoint six decades earlier than the power range channel, but it does not have to complete the trip action until the same time as the power range to provide the same level of protection. TVA has estimated this time to be approximately 0.5 seconds (see attached figure). Attached is a marked-up technical specification page that adds a 1.0 second response time for the source range channels.

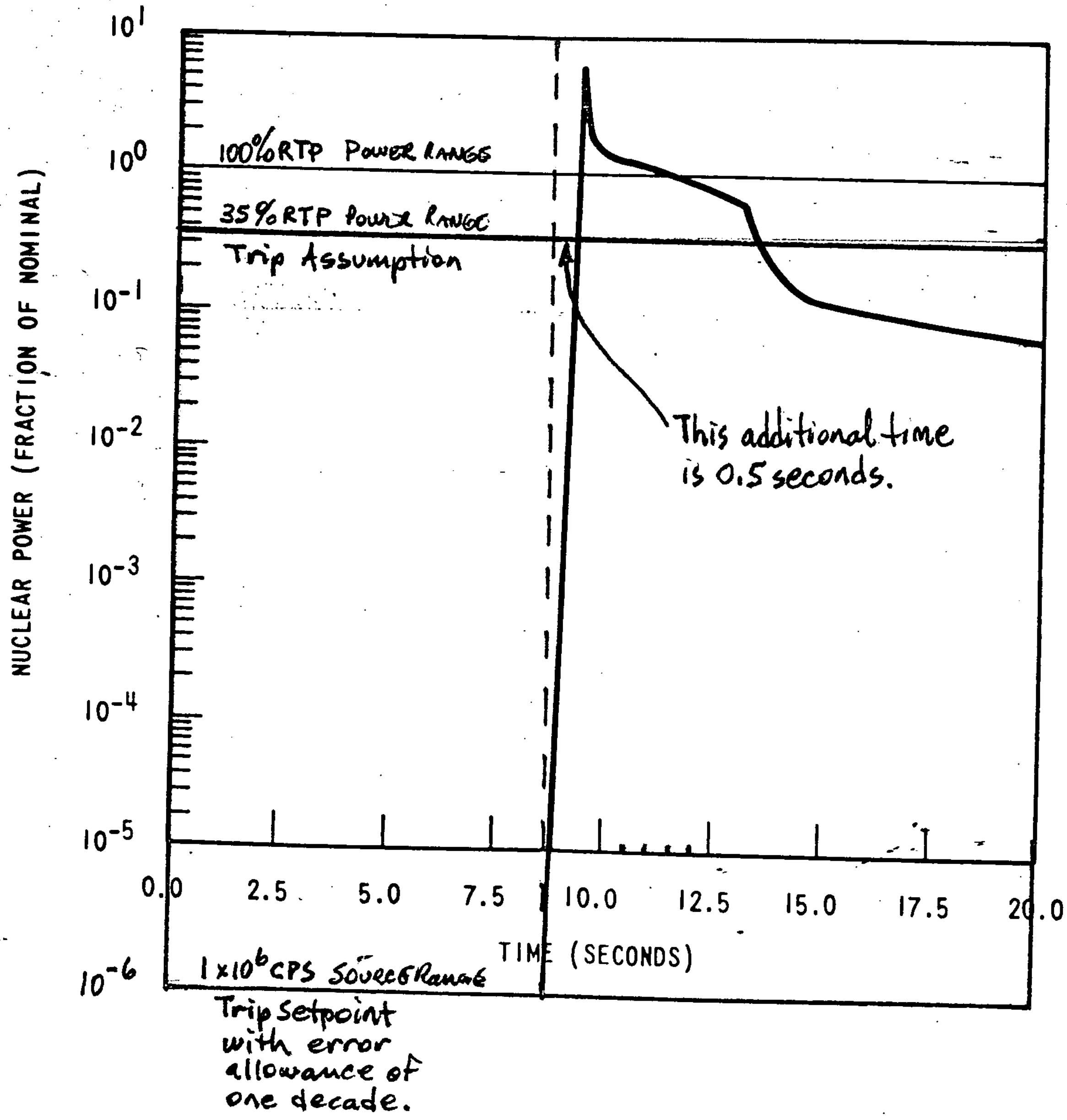

Figure 15.2-1 Uncontrolled Rod Withdrawal from a Subcritical Condition Nuclear Power Versus Time

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	≤ 1 second*
7. Overtemperature ΔT	≤ 4 seconds*
8. Overpower ΔT	≤ 4 seconds*
9. Pressurizer Pressure--Low	≤ 2 seconds
10. Pressurizer Pressure--High	≤ 2 seconds
11. Pressurizer Water Level--High	N.A.

FINAL DRAFT

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

MSIV Handswitches

References: 1) Letter from E. G. Adensam to H. G. Parris dated August 22, 1984

2) Letter from J. A. Domer to E. Adensam dated January 3, 1985

3) Letter from T. M. Novak to H. G. Parris dated April 3, 1985

In reference 1 NRC questioned the need for main steam isolation valve (MSIV) manual actuation capability to mitigate a steam generator tube rupture (SGTR) event in mode 4 (question 4.d). TVA responded that this issue was being addressed by the generic resolution of SGTR questions by a subgroup of the Westinghouse Owners Group in which TVA is participating. NRC accepted this approach in supplement 3 to the safety evaluation report and condition 18 to the license in the draft provided to TVA by reference 3. However, TVA understands from discussions with the reviewers that it is NRC's position that this issue must be addressed for fuel load to avoid any material false statements. TVA believes that this approach is inconsistent with the generic resolution of SGTR questions. But to avoid any material false statements and delays in the licensing of Watts Bar, TVA is providing a marked-up technical specification page to require MSIV manual actuation in mode 4. If TVA's understanding of this problem is in error, TVA prefers to delay implementation of this change until generic resolution of the SGTR questions in order that it be properly evaluated in the context of the entire issue.

0195K

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
c. Containment Ventilation Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4. Steam Line Isolation					
a. Manual Initiation	1/steam line	1/steam line	1/operating steam line	1, 2, 3,4	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Containment Pressure--High-High (P-14)	4	2	3	1, 2, 3	16
d. Steam Flow in Two Steam Lines--High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3	15*
Coincident With Either T_{avg}--Low-Low	4 (1 T_{avg}/loop)	2	3	1, 2, 3	15*
Or Steam Line Pressure-Low	4 (1 pressure/loop)	2	3	1, 2, 3	15*

FINAL DRAFT

FEB 15 1985

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
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3,4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident With	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
Either T_{avg}--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
Or Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

FINAL DRAFT

Technical Specification 4.5.2.h.1 - Charging Pump Flow

Westinghouse (W) has performed a revised ECCS analysis to include the CCP miniflow lines being in the open position on an SI signal. We have received the revised 'n-1' injection flow rate of 3150 gpm from Westinghouse via their site representative. Westinghouse formal report will follow shortly.

CARBON TO PERSON ADDRESSED. TRAFFORD, PA. AND RETURN PINK COPY TO SENDER.

TO:		FROM:
M.K Jones	Contract # 71C62-54114-1	G.W. Yetter
Engineering Supervisor		NOD/IS Site Manager
TVA Watts Bar Site		TVA Watts Bar Site

SUBJECT

Centrifugal Charging Pump Mini Flow.

DATE

04/08/85

Per telecon 4/8/85 Yetter/Dudak, the verified mini flow rate for Unit #1 (WAT) pump is 315.0 GPM. This flow rate is acceptable with the mini flow valves open or closed.

I have been assured the figure 315.0 GPM is acceptable to both ECCS and Containment Analysis groups at Westinghouse.

SIGNATURE

REPLY

FORM 31522C SIGNATURE DATE

Recipient

Engineered Safety Features Response Times

Table 3.3-5, items 2.a.4 and 3.a.4

The containment ventilation isolation response time of 5.5 seconds is based on loss of coolant accident (LOCA) considerations. Containment purge valves are assumed to be open at the start of the accident and assumed to close in 5.5 seconds. The response time is necessary to limit the air lost from the containment because of the influence of containment backpressure on reflood rates. The response time is conposed of 4 seconds for the purge valves and 1.5 seconds for signal generation. The 4 second valve time is listed in Table 3.6-2 for the purge valves. The proposed note (11) is necessary to clarify that the response time of 5.5 seconds is applicable only to the purge valves.

The upper and lower containment radiation monitor valves also close on a containment ventilation isolation signal. However, the radiation monitor system is a closed system. Prompt operation of these valves is not necessary to assure the validity of the LOCA analysis. They have a 5 second stroke time listed in Table 3.6-2. The signal generation time is the same (1.5 seconds). The proper response time for these valves would be 6.5 seconds. The proposed note (11) clarifies that the containment ventilation response time listed applies to the purge valves only.

FINAL DRAFT

FEB 15 1985

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Ventilation Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Raw Cooling Water	N.A.
j. Control Room Isolation	N.A.
k. Containment Air Return Fan	N.A.
l. Component Cooling Water	N.A.
m. Start Diesel Generators	N.A.
n. Reactor Trip	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 27(1)/12(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8(3)
3) Containment Isolation-Phase "A"(6)	≤ 18(2)/28(1)
4) Containment Ventilation Isolation	≤ 5.5(2)(11)
5) Auxiliary Feedwater Pumps	≤ 60(10)
6) Essential Raw Cooling Water	≤ 65(2)/75(1)
7) Control Room Isolation	N.A.
8) Component Cooling Water	≤ 43(2)/45(1)
9) Start Diesel Generators	≤ 12
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27(1)/12(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8(3)
3) Containment Isolation-Phase "A"(6)	≤ 18(2)/28(1)
4) Containment Ventilation Isolation	≤ 5.5(2)(11)

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FEB 15 1985

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air operated valves.

(4) Diesel generator starting and sequence loading delay included. RHR & SI pumps not included.

(5) Diesel generator starting and sequence loading delays not included. SI and RHR pumps not included.

(6) The following valves are exceptions to the response time shown in the table and will have the following response times for the initiating signals and functions:

FCV-70-143		FCV-62-77 and FCV-26-240, -243		FCV-61-96, -97, -110, -122, -191, -192, -193, -194	
2.a.3	$62^{(2)}/72^{(1)}$	2.a.3	$22^{(2)}/32^{(1)}$	2.a.3	32
3.a.3	$62^{(2)}/72^{(1)}$	3.a.3	$22^{(2)}/32^{(1)}$	3.a.3	32
4.a.3	$62^{(2)}/72^{(1)}$	4.a.3	$22^{(2)}/32^{(1)}$	4.a.3	32
5.a.3	$64^{(2)}/74^{(1)}$	5.a.3	$24^{(2)}/34^{(1)}$	5.a.3	34
6.a.3	$62^{(2)}/72^{(1)}$	6.a.3	$22^{(2)}/32^{(1)}$	6.a.3	32

(8) On 2/3 in 2/4 steam generators.

(9) The response time is measured from the time the 6.9 kV shutdown boards voltage exceeds the Setpoint until the time full voltage is returned for the loss of voltage sensors; or from the time the degraded voltage timers generate a signal to start the diesels or shed loads until the time full voltage is returned for the degraded voltage sensors.

(10) For motor-driven pumps only, the diesel generator starting and sequence loading delays are included.

(11) Containment purge valves only.

Technical Specification 4.7.7.a - MCR Temperature

TVA is presently researching its procurement documents to ensure that the 104°F air temperature is consistent with equipment qualification. TVA will notify NRC promptly if our research determines the current temperature limit must be revised.

Technical Specification 3.4.1.2

TVA has been informed by Westinghouse that its current RCCA withdrawal analysis assumes four reactor coolant pumps (RCPs) in operation. Westinghouse is presently performing a revised analysis which assumes only 2 RCPs operating.

Typographical Errors

FINAL DRAFT

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

its

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 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% $\Delta k/k$ 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 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 charging pump OPERABILITY requirements with MODE change.

Natural Circulation Testing

In a May 30, 1984, letter from L. M. Mills to E. Adensam, TVA requested certain Technical Specification requirements be waived during the performance of natural circulation testing. The waiver of these requirements must be made for TVA to be able to certify the final draft of the Watts Bar Technical Specifications.

A27 840530 005

400 Chestnut Street Tower II

May 30, 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 Tennessee Valley Authority	))	Docket Nos. 50-390

Please refer to (1) my letter to L. S. Rubenstein dated April 9, 1980 which provided information on the Sequoyah Nuclear Plant (SQN) unit 1 low power test program as requested by Supplement No. 1 to the SQN Safety Evaluation Report (NUREG-0011), and (2) H. R. Denton's letter to H. G. Parris dated July 10, 1980, which issued Amendment No. 4 to License No. DPR-77 (SQN unit 1) concerning the subject low power test program.

TVA plans to perform one type of natural circulation test several times during the Watts Bar Nuclear Plant (WBN) unit 1 startup test program for operator training.

The applicability of the Technical Specification (TS) safety limit, figure 2.1-1 of the TS, should be waived during performance of the natural circulation tests. This figure is based on four reactor coolant pumps in operation. During performance of the tests, no reactor coolant pumps will be in operation.

During performance of the tests, the overpower and overtemperature delta-T trip functions will be considered inoperable. These trip functions obtain temperature inputs from sensors located in the resistance temperature detector bypass loops. During natural circulation, the bypass loop flow will be extremely low causing the temperature indication to be in error and the response time characteristics to be slowed. The TS requirement 2.2.1, items 7 and 8, should be waived during performance of these tests.

TVA plans to isolate the Upper Head Injection (UHI) system during performance of these tests. This will be done to prevent inadvertent actuation of the system and the potential for economic damage to the reactor internals. The UHI system provides borated water to mitigate the consequences of a large loss of coolant accident. Evaluations done for the SQN natural circulation test program established that this system provides little or no benefit for accidents involving low power or decay heat levels. TS requirement 3.5.1.2 should be waived during performance of these tests.

Director of Nuclear Reactor Regulation May 30, 1984

Please ensure that the WBN unit 1 low power license contains the requested exemptions to the TS for the purpose of performing the natural circulation tests. By the previously referenced amendment to the SQN license, NRC granted similar exemptions that were requested by TVA.

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

L. M. Mills, Manager
Nuclear Licensing

Sworn to and subscribed before me
this 30th day of May 1984
Paulette H. White
Notary Public
My Commission Expires 9-5-84

cc: U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

RHS:DBE:LHB
cc: ARMS, 640 CST2-C
H. L. Abercrombie, 1750 CST2-C
J. W. Anderson, 255 SPB-K
E. A. Belvin, 109 MPB-M
T. G. Campbell, 1750 CST2-C
H. N. Culver, 249A HBB-K
G. W. Killian, 401 UBB-C (2)
J. A. Raulston, W10C126 C-K
H. S. Sanger, Jr., E11B33 C-K
M. Shymlock, Watts Bar-NRC
F. A. Szczepanski, 220 401B-C

COORDINATED: Memo from Coffey to Mills dated 5/3/84 (L33 840427 818).

TECHNICAL SPECIFICATION TABLE 2.2-1

In a January 30, 1985 letter to NRC, TVA requested changes to Table 2.2-1 to include allowances provided in the setpoint study which had been previously submitted to NRC. One change to item 14 of Table 2.2-1 was not included in the requested changes. The allowable value for steam generator water level at 100% of nominal load should be 52.9% of narrow range span. This same change was submitted and approved by NRC for the allowable value for steam generator water level in item 15 of Table 2.2-1. The same component provides the trip setpoint for item 14 and the steam generator water level portion of item 15; therefore, the allowable setpoint must be consistent. Attached is a marked-up page showing the necessary change.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
14. Steam Generator Water Level--Low-Low	≥ 17% of narrow range span between 0 and 35% load, increasing linearly to ≥ 54.9% of narrow range span at 100% of nominal load	≥ 15.0% of narrow range span between 0 and 35% load increasing linearly to ≥ ~~53.5%~~ 52.9% of narrow range span at 100% of nominal load
15. Steam Generator Water Level-Low Coincident With Steam/Feedwater Flow Mismatch	≤ 38% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 17% narrow range span between 0 and 35% load, increasing linearly to ≥ 54.9% of narrow range span at 100% of nominal load	≤ 41.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 15.0% of narrow range span between 0 and 35% load, increasing linearly to 52.9% of narrow range span at 100% of nominal load
16. Undervoltage-Reactor Coolant Pumps	≥ 4830 volts-each bus	≥ 4761 volts-each bus
17. Underfrequency-Reactor Coolant Pumps	≥ 57 Hz - each bus	≥ 56.9 Hz - each bus
18. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 43 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
19. Safety Injection Input from ESF	N.A.	N.A.

FINAL DRAFT

FEB 15 1985

Nominal Voltage Setpoint for Undervoltage on the 6.9-kV Shutdown Board

Technical Specification Table 3.3-4

In a letter to NRC dated September 14, 1984, TVA submitted the nominal voltage setpoints for undervoltage on the 6.9-kV shutdown boards. The value submitted, 4860 volts, was taken from the Sequoyah Nuclear Plant Technical Specifications because it was believed that the undervoltage relays at Sequoyah were the same as at Watts Bar. The undervoltage relays at Watts Bar are set at 4830 volts and the trip setpoint in Table 3.3-4 should be 4830 volts. The allowable value should be 4830 volts with a tolerance of plus or minus 96.6 volts which corresponds to plus or minus 2 percent of the allowable value. Attached are marked-up technical specification pages showing the changes to items 6.e.1, 8.a.1.a, and 8.a.2.a of Table 3.3-4.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	$\leq$ 82.4% of narrow range instrument span each steam generator	$\leq$ 84.4% of narrow range instrument span each steam generator
6. Auxiliary Feedwater		
a. Manual Initiation	N. A.	N. A.
b. Automatic Actuation Logic and Actuation Relays	N. A.	N. A.
c. Steam Generator Water Level-Low-Low Start Motor-Driven Pumps and Turbine-Driven Pump	$\geq$ 17% of narrow range instrument span between 0 and 35% load increasing linearly to $\geq$ 54.9% of narrow range span at 100% nominal load	$\geq$15.0% of narrow range instrument span between 0 and 35% load increasing linearly to $\geq$ 52.9% of narrow range span at 100% nominal load
d. Safety Injection Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints/ Allowable Values.	
e. Loss-of-Offsite Power-Start Motor-Driven Pumps Start Turbine-Driven Pump		
1) Nominal Voltage Setpoint	4830 ~~4860~~ volts	4830 ~~4860~~ ± 96.6 ~~97.2~~ volts
2) Relay Response Time	0.0 volt input to the inverse time relay with a 5 second time delay	0.0 volt input to the inverse time relay with a 5 ± 1 second time delay
f. Trip of All Main Feedwater Pumps - Start Motor-Driven Pumps and Turbine-Driven Pump	N.A.	N.A.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. Auxiliary Feedwater (continued)		
g. Auxiliary Feedwater Suction Pressure-Low (Suction Transfer to ERCW)		
1) Supply Valve for Motor-Driven Pump	$\geq$ 1.70 psig	$\geq$ 0.95 psig
2) Supply Valve for Turbine-Driven Pump	$\geq$ 11.1 psig	$\geq$ 10.0 psig
7. Automatic Switchover To Containment Sump		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. RWST Level - Low Coincident With	$\geq$ 130" from tank base	$\geq$ 126" from tank base
Containment Sump Level - High	$\geq$ 30" above elev. 703'	$\geq$ 32.5" above elev. 703'
And		
Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints/Allowable Values.	
8. 6.9 kV Shutdown Board		
a. Loss of Voltage		
1) Start Diesel Generator		
a) Nominal Voltage Setpoint	4830 ~~4860~~ volts	4830 ~~4860~~ ± 96.6 ~~97.2~~ volts
b) Relay Response Time	0.0 volt input to the inverse time relay with a 1.5 second time delay	0.0 volt input to the inverse time relay with a 1.5 ± 0.5 second time delay

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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 (continued)		
2) Load Shedding a) Nominal Voltage Setpoint	~~4860~~ 4830 volts	~~4860~~ 4830 ± ~~97.2~~ 96.6 volts
b) Relay Response Time	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 volts	6560 ± 33 volts
2) Diesel Generator Start and Load Shedding Timer	300 seconds	300 ± 30 seconds
3) Safety Injection Degraded Voltage Logic Enable Timer	10 seconds	10 ± 1 seconds
9. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	$\leq$ 1970 psig	$\leq$ 1980 psig
b. Low-Low T_{avg}, P-12, increasing decreasing	$\geq$ 550°F $\leq$ 550°F	$\leq$ 552°F $\geq$ 548°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Water Level, P-14	See Item 5. above for all Steam Generator Water Level-High-High Trip Setpoints and Allowable Values.	

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ENGINEERED SAFETY FEATURES RESPONSE TIME

Table 3.3-5

In the Proof and Review copy of the technical specifications the response time for steam line isolation on high steam flow in two steam lines coincident with low average reactor coolant temperature (item 5.6) is 9 seconds. The response time apparently was changed by NRC to 7 seconds in the final draft copy of the technical specifications. TVA did not request this change. The valve stroke time listed in specificaion 3.7.1.5 is 5 seconds. The additional channel response time to generate an isolation signal for a channel involving temperature measurement is 4 seconds. The total response time is the sum of the two times or 9 seconds. The temperature measuring channels are 2 seconds slower than the normal channel because of the RTD response time. Hence the response time for item 5.b should be 2 seconds slower than item 6.b in table 3.3-5.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low (Continued)	
5) Auxiliary Feedwater Pumps	$\leq 60^{(10)}$
6) Essential Raw Cooling Water	$\leq 65^{(2)}/75^{(1)}$
7) Control Room Isolation	N.A.
8) Component Cooling Water	$\leq 43^{(2)}/45^{(1)}$
9) Start Diesel Generators	≤ 12
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 22^{(4)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 8^{(3)}$
3) Containment Isolation-Phase "A"(6)	$\leq 18^{(2)}/28^{(1)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater Pumps	$\leq 60^{(10)}$
6) Essential Raw Cooling Water	$\leq 65^{(2)}/75^{(1)}$
7) Control Room Isolation	N.A.
8) Component Cooling Water	$\leq 43^{(2)}/45^{(1)}$
9) Start Diesel Generators	≤ 12
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg} LowLow	
a. Safety Injection (ECCS)	$\leq 24^{(4)}/14^{(5)}$
1) Reactor Trip (from SI)	≤ 4
2) Feedwater Isolation	$\leq 10^{(3)}$
3) Containment Isolation-Phase "A"(6)	$\leq 20^{(2)}/30^{(1)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater Pumps	$\leq 60^{(10)}$
6) Essential Raw Cooling Water	$\leq 67^{(2)}/77^{(1)}$
7) Control Room Isolation	N.A.
8) Component Cooling Water	$\leq 43^{(2)}/45^{(1)}$
9) Start Diesel Generators	≤ 14
b. Steam Line Isolation	≤ 79

Tables 3.3-6 and 4.3-3

By letter dated February 16, 1985, TVA requested an exemption to 10 CFR 70.74 for having a criticality monitor. It is anticipated that this request will be granted since the draft license forwarded to TVA by letter dated March 4, 1985, included this as an exemption. Therefore, the subject tables should be revised as indicated to delete the reference to a criticality monitor.

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. Auxiliary Building Isolation					
Radiation Level-High ~~and Criticality~~ (RE-90-102 and RE-90-103)	1	2	*	$\leq$ 15 mR/h	28
2. 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
3. Control Room Ventilation Isolation					
Control Room Air Intake Radioactivity-High (RE-90-125 and RE-90-126)	1	2	All	400 cpm**	27

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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. Auxiliary Building Isolation				
Radiation Level-High ~~and Criticality~~ (RE-90-102 and RE-90-103)	S	R	M	*
2. 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
3. Control Room Ventilation Isolation				
Control Room Air Intake Radioactivity-High (RE-90-125 and RE-90-126)	S	R	M	All

* - With fuel in the fuel storage areas.

SEISMIC MONITORING INSTRUMENTATION

TECHNICAL SPECIFICATION TABLE 3.3-7

The measurement range for the triaxial peak accelerographs described in items 2.b and 2.c should be 0 to 2.0g. Attached is a marked up technical specification page showing this change. This change must be made for this technical specification to accurately reflect the as-built plant.

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. 0-XT-52-75A Annulus El. 703	0 - 1.0g	1*
b. 0-XT-52-75B Cont. El. 757	0 - 1.0g	1*
c. 0-XT-52-75D D/G Bldg. El. 742	0 - 1.0g	1*
2. Triaxial Peak Accelerographs		
a. 0-XR-52-76A Cont. El. 725	0 - 5.0 g	1
b. 0-XR-52-76B Cont. El. 730	0 - ~~5~~2.0 g	1
c. 0-XR-52-76D Control Bldg. El. 755	0 - ~~5~~2.0 g	1
3. Triaxial Seismic Switches		
0-XS-52-80 Annulus El. 703	0.025 - 0.25g	1*
4. Triaxial Response-Spectrum Recorders		
a. 0-XR-52-77A Annulus El. 703	2 - 25.4 Hz	1*
b. 0-XR-52-77B Cont. El. 757	2 - 25.4 Hz	1
c. 0-XR-52-77D Cont. El. 755	2 - 25.4 Hz	1
d. 0-XR-52-77E D/G Bldg. El. 742	2 - 25.4 Hz	1

*With reactor control room indication

ENGINEERED SAFETY FEATURES RESPONSE TIMES

Table 3.3-5

Table notation (6) should be revised to reflect the correct valve response time. The correct valve stroke time of 66 seconds is listed in Table 3.6-2. The channel response time of 2 seconds (4 seconds for channels with temperature measurement) and 10 seconds for diesel response time must also be included in the overall engineered safety feature response time. This information was previously transmitted to NRC in a letter from D. L. Lambert to E. Adensam dated January 30, 1985.

Table notation (7) was apparently omitted from the February 15, 1985, certification version of the technical specifications when table notation (6) was revised. It should be reinserted.

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TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air operated valves.

(4) Diesel generator starting and sequence loading delay included. RHR & SI pumps not included.

(5) Diesel generator starting and sequence loading delays not included. SI and RHR pumps not included.

(6) The following valves are exceptions to the response time shown in the table and will have the following response times for the initiating signals and functions:

FCV-70-143		FCV-62-77 and FCV-26-240, -243		FCV-61-96, -97, -110, -122, -191, -192, -193, -194	
2.a.3	68 ~~62~~(2)/~~72~~(1) 78	2.a.3	22(2)/32(1)	2.a.3	32
3.a.3	68 ~~62~~(2)/~~72~~(1) 78	3.a.3	22(2)/32(1)	3.a.3	32
4.a.3	68 ~~62~~(2)/~~72~~(1) 78	4.a.3	22(2)/32(1)	4.a.3	32
5.a.3	70 ~~64~~(2)/~~74~~(1) 80	5.a.3	24(2)/34(1)	5.a.3	34
6.a.3	68 ~~62~~(2)/~~72~~(1) 78	6.a.3	22(2)/32(1)	6.a.3	32

(8) On 2/3 in 2/4 steam generators.

(9) The response time is measured from the time the 6.9 kV shutdown boards voltage exceeds the Setpoint until the time full voltage is returned for the loss of voltage sensors; or from the time the degraded voltage timers generate a signal to start the diesels or shed loads until the time full voltage is returned for the degraded voltage sensors.

(10) For motor-driven pumps only, the diesel generator starting and sequence loading delays are included.

(7) On 2/3 any steam generator

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Table 4.3-2

The modes for which surveillance is required for item 1.c should include mode 4 to make it consistent with Table 3.3-3.

TABLE 4.3-2

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
1. Safety Injection (Reactor Trip, Turbine Trip, Feedwater, Isolation, Control Room Isolation, Start Diesel Generators, Component Cooling Water, and Essential Raw Cooling Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident With	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
Either 1) T_{avg}--Low-Low Or	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2) Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Table 4.3-2

Table notation (4) should be revised to be consistent with the definitions presented in table 1.1.

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TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Monthly testing shall consist of relay testing excluding final actuation of the pumps or valves.

(3) Monthly testing shall consist of voltage sensor relay testing excluding actuation of the load shedding, diesel start, and time-delay timers.

(4) Slave relays K603A, K603B, K604A, K604B (SI) and K625A, K625B (Phase B isolation) shall be tested during each cold shutdown exceeding 24 hours unless tested during the previous ~~3 months~~ 92 days. K609A, K609B (SI) shall be tested every 18 months.

Technical Specification Table 3.3-6

ACTION Statement 28 references ACTION b. of Specification 3.9.12 and adds a requirement to have one ABGTS train in operation. This additional requirement is unnecessary since all fuel movement activities will be stopped. The additional requirement is also inconsistent with the wording in revision 5 of the Standard Technical Specifications (NUREG-0452).

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TABLE 3.3-6 (Continued)

TABLE NOTATIONS

*With fuel in the fuel storage areas.

**400 cpm is equivalent to 1×10^{-5} $\mu Ci/cm^3$ of Xe-133.

ACTION STATEMENTS

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 initiate operation of the Control Room Ventilation System in the recirculation mode.

ACTION 28 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, ACTION a. of Specification 3.9.12 must be satisfied. With both channels inoperable, provide an appropriate portable continuous monitor with the same Alarm Setpoint in the fuel pool area and satisfy ACTION b. of Specification 3.9.12 ~~with one Auxiliary Building Gas Treatment System train in operation~~.

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

SEISMIC MONITORING INSTRUMENTATION

Tables 3.3-7 and 4.3-4

The correct location for triaxial response spectrum recorder 0-XR-52-77D is the Auxiliary Control Room, elevation 757.

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. 0-XT-52-75A Annulus El. 703	0 - 1.0g	1*
b. 0-XT-52-75B Cont. El. 757	0 - 1.0g	1*
c. 0-XT-52-75D D/G Bldg. El. 742	0 - 1.0g	1*
2. Triaxial Peak Accelerographs		
a. 0-XR-52-76A Cont. El. 725	0 - 5.0 g	1
b. 0-XR-52-76B Cont. El. 730	0 - 5.0 g	1
c. 0-XR-52-76D Control Bldg. El. 755	0 - 5.0 g	1
3. Triaxial Seismic Switches		
0-XS-52-80 Annulus El. 703	0.025 - 0.25g	1*
4. Triaxial Response-Spectrum Recorders		
a. 0-XR-52-77A Annulus El. 703	2 - 25.4 Hz	1*
b. 0-XR-52-77B Cont. El. 757	2 - 25.4 Hz	1
c. 0-XR-52-77D Aux. Cont. Room El. ~~755~~ 757	2 - 25.4 Hz	1
d. 0-XR-52-77E D/G Bldg. El. 742	2 - 25.4 Hz	1

*With reactor control room indication

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Time-History Accelerographs			
a. 0-XT-52-75A Annulus El. 703**	M*	R***	SA
b. 0-XT-52-75B Cont. El. 757**	M*	R***	SA
c. 0-XT-52-75D D/G Bldg. El. 742**	M*	R***	SA
2. Triaxial Peak Accelerographs			
a. 0-XR-52-76A Cont. El. 725	N.A.	R	N.A.
b. 0-XR-52-76B Cont. El. 730	N.A.	R	N.A.
c. 0-XR-52-76D Control Bldg. El. 755	N.A.	R	N.A.
3. Triaxial Seismic Switches			
0-XS-52-80 Annulus El. 703**	M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. 0-XR-52-77A Annulus El. 703**	M	R	SA
b. 0-XR-52-77B Cont. El. 757	N.A.	R	N.A.
c. 0-XR-52-77D Cont. El. ~~755~~ 757	N.A.	R	N.A.
d. 0-XR-52-77E D/G Bldg. El. 742	N.A.	R	N.A.

Aux. Room

*Except seismic trigger.
**With reactor control room indications.
***Includes seismic trigger.

ACCIDENT MONITORING INSTRUMENTATION

Table 4.3-7

The radiation monitor numbers should be added to Table 4.3-7 to make it consistent with Table 3.3-10.

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
18. Shield Building Vent-High Range Noble Gas Monitor (RE-90-401)	M	R*
19. Condenser Vacuum Exhaust Vent-High Range Noble Gas Monitor (RE-90-404)	M	R*
20. Steam Line Relief-Noble Gas Monitor (RE-90-421, 422, 423, and 424)	M	R*
21. Reactor Vessel Water Level**	M	R
22. Containment Atmosphere - High Range Monitor (RE-90-271, 272, 273, and 274)	M	R*

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

**The surveillance requirements for the Reactor Vessel Water Level system are not applicable until startup following the first refueling outage.

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FIRE DETECTOR INSTRUMENTATION

Table 3.3-11

Zone 225 on page 3/4 3-67 should require 11 function A smoke detectors.

Zones 56 through 59 and 64 through 65 on page 3/4 3-68 should be battery board rooms, not battery rooms.

Zones 41 and 42 on page 3/4 3-68 should be deleted. These zones contain only unit 2 equipment and the rooms are three hour rated enclosures.

Zones 45 and 46 on page 3/4 3-69 should be deleted. These zones contain only unit 2 equipment and the rooms are three hour rated enclosures.

Zones 120 and 121 on page 3/4 3-70 should require one function A smoke detector and no function B detectors, rather than the opposite.

Zones 123 and 125 on page 3/4 3-70 should have 3 function B detectors. Additional detectors were added as a result of the Appendix R inspections.

Zones 128 and 129 should be added to page 3/4 3-70 as a result of the Appendix R inspections.

Zones 455, 456, 330, 332, and 333 should be added to page 3/4 3-74 as a result of the Appendix R inspection.

Zones 457 and 458 should be added to page 3/4 3-75 as a result of the Appendix R inspections.

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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)			
214	Mech. Equip. Rm., Col. C1-C2, El. 755			0/5
215	Mech. Equip. Rm., Col. C1-C2, El. 755			0/5
216	CR Fltr. B, Duct Det., El. 755			0/1
217	CR Fltr. B, Duct Det., El. 755			0/1
218	CR Fltr. A, Duct Det., El. 755			0/1
219	CR Fltr. A, Duct Det., 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			11/0 ~~12/0~~
227	Operation Living Area, El. 755			0/8
228	Operation Living Area, El. 755			0/8
267	Aux. Instr. Rm., Unit 1, El. 708			0/8
268	Aux. Instr. Rm., Unit 1, El. 708			0/10
269	Computer Rm., El. 708			0/4
270	Computer Rm., El. 708			0/4
271	Aux. Instr. Rm., Unit 2, El. 708			0/8
272	Aux. Instr. Rm., Unit 2, El. 708			0/10
273	Computer Rm. Corridor, El. 708			3/0

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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)
B.	Control Building (Continued)			
298	Common Main Cont. Boards & M-15, El. 755			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
54	Battery Rm., El. 692			0/3
55	Battery Rm., El. 692			0/3
56	Battery Board Rm., El. 692			2/0
57	Battery Board Rm., El. 692			2/0
58	Battery Board Rm., El. 692			2/0
59	Battery Board Rm., El. 692			2/0
60	Battery Rm., El. 692			0/3
61	Battery Rm., El. 692			0/3
62	Battery Rm., El. 692			0/3
63	Battery Rm., El. 692			0/3
64	Battery Board Rm., El. 692			2/0
65	Battery Board Rm., El. 692			2/0
387	Control/Turbine Bldg. Wall	0/26		
C.	Auxiliary Building			
39	Cont. Spray Pump 1A-A, El. 676			2/0
40	Cont. Spray Pump 1B-B, El. 676			2/0
~~41~~	~~Cont. Spray Pump 2A-A, El. 676~~			~~2/0~~
~~42~~	~~Cont. Spray Pump 2B-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)			
43	RHR Pump 1A-A, El. 676			2/0
44	RHR Pump 1B-B, El. 676			2/0
~~45~~	~~RHR Pump 2A-A, El. 676~~			~~2/0~~
~~46~~	~~RHR Pump 2B-B, El. 676~~			~~2/0~~
47	Corridor of Aux. Bldg., El. 676			11/0
70	A5-A11, Col. W-X, El. 692			0/5
71	A5-A11, Col. W-X, El. 692			0/5
72	Aux. FW Pump Turbine 1A-S, El. 692			0/1
73	Aux. FW Pump Turbine 1A-S, El. 692			0/1
76	S.I. & Charging Pump Rms., El. 692			0/5
77	S.I. Pump Rm. 1A, El. 692			0/1
78	S.I. Pump Rm. 1B, El. 692			0/1
79	Charging Pump Rm. 1C, El. 692			0/1
80	Charging Pump Rm. 1B, El. 692			0/1
81	Charging Pump Rm. 1A, El. 692			0/1
88	Aux. Bldg. Corridor A1-A8, El. 692			0/8
89	Aux. Bldg. Corridor A1-A8, El. 692			0/8
90	Aux. Bldg. Corridor A8-A15, El. 692			0/8
91	Aux. Bldg. Corridor A8-A15, El. 692			0/8
92	Aux. Bldg. Corridor Col. U-W, El. 692			0/4
93	Aux. Bldg. Corridor Col. U-W, El. 692			0/4
94	Pipe Gallery, El. 692			0/2
95	Pipe Gallery, El. 692			0/2

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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., A & B, Duct. Det., El., 713			0/2
99	Cntmt. Purge Air Fltr., A & B, Duct. Det., 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
108	Radio Chemical Lab. Area, El. 713			0/3
109	Radio Chemical Lab. Area, El. 713			0/3
110	Aux. Bldg. A1-A8, Col. Q-U, El. 713			0/18
111	Aux. Bldg. A1-A8, Col. Q-U, El. 713			0/19
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
116	Cask Loading Area, El. 729			0/2
117	Cask Loading Area, El. 729			0/2
118	New Fuel Storage Area			4/0
120	Aux. Bldg. Gas Trtmt. Fltr., El. 737			~~0/1~~ 1/0
121	Aux. Bldg. Gas Trtmt. Fltr., El. 737			~~0/1~~ 1/0
123	Vol. Control Tank Rm. 1A, El. 713			~~0/1~~ 0/3
125	Vol. Control Tank Rm. 1A, El. 713			~~0/1~~ 0/3
128	Post Accident Samp Fac U-1, El. 729			0/3
129	Post Accident Samp Fac U-1, El. 729			0/3

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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)			
242	480V XFMR Rm. 1A, El. 772			0/3
243	480V XFMR Rm. 1B, El. 772			0/3
244	480V XFMR Rm. 1B, El. 772			0/3
245	480V XFMR Rm. 2A, El. 772			0/3
246	480V XFMR Rm. 2A, El. 772			0/3
247	480V XFMR Rm. 2B, El. 772			0/3
248	480V XFMR Rm. 2B, El. 772			0/3
249	125V Batt. Rm. I, El. 772			2/0
251	125V Batt. Rm. II, El. 772			2/0
253	125V Batt. Rm. III, El. 772			2/0
255	125V Batt. Rm. IV, El. 772			2/0
257	480V Bd. Rm. 1B, El. 772			0/4
258	480V Bd. Rm. 1B, El. 772			0/4
259	480V Bd. Rm. 1A, El. 772			0/4
260	480V Bd. Rm. 1A, El. 772			0/4
261	480V Bd. Rm. 2A, El. 772			0/4
262	480V Bd. Rm. 2A, El. 772			0/4
263	480V Bd. Rm. 2B, El. 772			0/4
264	480V Bd. Rm. 2B, El, 772			0/4

INSERT

		SMOKE
455	Post Acc Samp fac, U-1, El 737	0/2
456	Post Acc Samp fac, U-1, El 737	0/2
		Smoke
330	Pipe Chase, U-1, El, 737, 713, 692	20/0
332	North Main Stm Vlv Rm el, 737	4/0
333	South Main Stm Vlv Rm el, 737	4/0

ZONE	INSTRUMENT LOCATION	HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
D.	Additional Equipment Building			
122	Add. Eqpt. Bldg., Unit 1, El. 729			6/0
154	Add. Eqpt. Bldg., Unit 1, El. 763.5			6/0
231	Add. Eqpt. Bldg., El. 786.5			4/0

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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)
D.	Additional Equipment Building (Continued)			
232	Add. Eqpt. Bldg., El. 775.25			4/0
E.	Intake Pumping Station			
250	ERCW Pmp. Rm., El. 741	4/0		
277	Strainer Rm., El. 722			18/0
278	ECRW Pmp. Rm., El 741	4/0		
405	Elect. Bd. Rm., El. 711			0/5
406	Elect. Bd. Rm., El. 711			0/5
F.	Containment#			
352	Lwr. Compt. Coolers, El. 716			4/0
354	Upr. Compt. Coolers, El. 801			4/0
356	RCP 2, El. 716		0/2	
357	RCP 2, El. 716	0/2		
360	RCP 1, El. 716		0/2	
361	RCP 1, El. 716	0/2		
364	RCP 3, El. 716		0/2	
365	RCP 3, El. 716	0/2		
368	RCP 4, El. 716		0/2	
369	RCP 4, El. 716	0/2		
372	Reactor Bldg. Annulus			0/20
373	Reactor Bldg. Annulus			0/19
G.	Additional Diesel Generator Building			
425	Add. D/G Rm., Fuel Trf. Rm. & Pipe Gallery	0/8		
426	Add. D/G Rm., Fuel Trf. Rm. & Pipe Gallery	0/8		

Smoke

457 Reactor Bldg. Annulus 0/9

458 Reactor Bldg. Annulus 0/8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Table 3.3-12

Item 3.b should be renumbered 3.c and item 3.c should be renumbered 3.b. This will make Table 3.3-12 consistent with Table 4.3-8, Watts Bar surveillance instructions reference the numbering in Table 4.3-8, so it should not be changed.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Waste Disposal System Liquid Effluent Line (RE-90-122)	1	31
b. Steam Generator Blowdown Effluent Line (RE-90-120 and 121)	1	32
c. Condensate Demineralizer Regenerant Effluent Line (RE-90-225)	1	31
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Essential Raw Cooling Water Effluent Line (RE-90-133 & 90-140 or RE-90-134 & 90-141)	1/discharge header	33
b. Turbine Building Sump Effluent Line (RE-90-212)	1	33
c. Plant Liquid Discharge Line (RE-90-211)	1	33
3. Flow Rate Measurement Devices		
a. Waste Disposal System Liquid Radwaste Effluent Line	1	34
c ~~b~~. Condensate Demineralizer Regenerant Effluent Line	1	34
b ~~c~~. Steam Generator Blowdown Effluent Line	1	34
d. Diffuser Discharge Effluent Line	1	34
4. Tank Level Indicating Devices		
a. Condensate Storage Tank	1	35
b. Steam Generator Layup Tank*	1	35

*Required when connected to the Secondary Coolant System.

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Table 4.3-8

A new note has been added for item 1.c. The three conditions listed in notes (1) and (2) provide alarm annunciation in the control room for radiation monitor RE-90-225. In addition, the effluent pathway is automatically isolated by one of the items, measured levels above the setpoint. Neither note (1) nor note (2) adequately address the design of RE-90-225. Therefore, a new note (5) has been added.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Waste Disposal System Liquid Effluent Line (RE-90-122)	D	P	R(3)	Q(1)
b. Steam Generator Blowdown Effluent Line (RE-90-120 and 121)	D	M	R(3)	Q(1)
c. Condensate Demineralizer Regenerant Effluent Line (RE-90-225)	D	M	R(3)	Q~~(1)~~ (5)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Essential Raw Cooling Water Effluent Line (RE-90-133 & 90-140 or RE-90-134 & 90-141)	D	M	R(3)	Q(2)
b. Turbine Building Sump Effluent Line (RE-90-212)	D	M	R(3)	Q(2)
c. Plant Liquid Discharge Line (RE-90-211)	D	M	R(3)	Q(2)
3. Flow Rate Measurement Devices				
a. Waste Disposal System Liquid Effluent Line	D(4)	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D(4)	N.A.	R	Q
c. Condensate Demineralizer Regenerant Effluent Line	D(4)	N.A.	R	Q
d. Diffuser Discharge Effluent Line	D(4)	N.A.	R	Q
4. Tank Level Indicating Devices				
a. Condensate Storage Tank	D*	N.A.	R	Q
b. Steam Generator Layup Tank	D*	N.A.	R	N.A.

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TABLE 4.3-8 (Continued)

TABLE NOTATIONS

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* During liquid additions to the tank.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

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

b. Circuit failure, or

c. Instrument indicates downscale failure.

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

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

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

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates downscale failure.

The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway occurs if the instrument indicates measured levels above the Alarm setpoint.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Tables 3.3-13 and 4.3-9

Radiation monitor RE-90-99 has been modified to have sensitivity capabilities equivalent to RE-90-119. Either monitor can be used to monitor normal effluent releases from the condenser vacuum exhaust pathway.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM (RE-90-118)			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	37
b. Effluent System Flow Rate Measuring Device	1	*	38
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitor	1	**	40
b. Oxygen Monitor	1	**	40
3. Condenser Vacuum Exhaust System (RE-90-119 or RE-90-99)			
a. Noble Gas Activity Monitor	1	*	39
b. Effluent System Flow Rate Measuring Device	1	*	38
c. Monitor Flow Rate Measuring Device	1	*	38
d. Iodine Sampler	1	****	41
e. Particulate Sampler	1	****	41
f. Sampler Flow Rate Measuring Device	1	****	38
4. Shield Building Exhaust System (RE-90-400)			
a. Noble Gas Activity Monitor	1	***	39
b. Iodine Sampler	1	***	41
c. Particulate Sampler	1	***	41
d. Effluent System Flow Rate Measuring Device	1	***	38
e. Sampler Flow Rate Measuring Device	1	***	38
f. Monitor Flow Rate Measuring Device	1	***	38

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM (RE-90-118)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
b. Effluent System Flow Rate Measuring Device	D	N.A.	R	Q	****
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System					
a. Hydrogen Monitor	D	N.A.	Q(4)	M	**
b. Oxygen Monitor	D	N.A.	Q(5)	M	**
3. Condenser Vacuum Exhaust System (RE-90-119 or RE-90-99)					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Effluent System Flow Rate Measuring Device	D	N.A.	R	Q	*
c. Monitor Flow Rate Measuring Device	D	N.A.	R	Q	*
d. Iodine Sampler	W	N.A.	N.A.	N.A.	*****
e. Particulate Sampler	W	N.A.	N.A.	N.A.	*****
f. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*****
4. Shield Building Exhaust System (RE-90-400)					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	***
b. Iodine Sampler	W	N.A.	N.A.	N.A.	***
c. Particulate Sampler	W	N.A.	N.A.	N.A.	***
d. Effluent System Flow Rate Measuring Device	D	N.A.	R	Q	***

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RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Table 4.3-9

The analog channel operational test notation has been changed from (1) to (2). The channel is designed to isolate the system and note (1) would seem to apply. However, note (1) requires that automatic isolation be demonstrated. This testing would require all purge valves to be opened and subsequently closed by the automatic isolation signal on a quarterly basis. Technical Specification 3.6.1.9 prohibits more than one purge line (supply and exhaust) open during power operation. The containment ventilation isolation actuations circuit is tested in accordance with the requirements in Technical Specification table 4.3-2, item 3.c. The containment purge valves are tested in accordance with the requirements of Technical Specification 3.6.3. The testing requirements in Table 4.3-9, item 7, should be made consistent with the other requirements in the Technical Specifications. The level of protection afforded by this change is the same as that provided for all containment isolation systems.

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Shield Building Exhaust System (Continued)					
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	***
f. Monitor Flow Rate Measuring Device	D	N.A.	R	Q	***
5. Auxiliary Building Ventilation And Fuel Handling Area Ventilation System (RE-90-101)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	M	R(3)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Measuring Device	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*
f. Monitor Flow Rate Measuring Device	D	N.A.	R	Q	*
6. Service Building Ventilation System (RE-90-132)					
a. Noble Gas Actvity Monitor	D	M	R(3)	Q(2)	*
b. Effluent System Flow Rate Measuring Device	D	N.A.	R	Q	*
c. Monitor Flow Rate Measuring Device	D	N.A.	R	Q	*
7. Containment Purge and Exhaust System (RE-90-130/131) Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	P	R(3)	Q(~~1~~) (2)	*

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TURBINE OVERSPEED PROTECTION

Surveillance Requirement 4.3.4.2

In a letter to NRC from J. A. Domer to E. Adensam dated 3/25/85 (L44 850325 811), TVA requested that the wording in surveillance requirement 4.3.4.2 be replaced with a reference to the Watts Bar Turbine Integrity Program with Turbine Overspeed Protection (TIPTOP) which would be described in section 6.8 of the technical specifications. This request must be resolved for certification of the technical specifications.

PRESSURE/TEMPERATURE LIMITS

Surveillance Requirement 4.4.9.1.2

The results of the specimen examinations should be used to update Figure 3.4-4, also.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, ~~and~~ 3.4-3, and 3.4-4

REACTOR COOLANT SYSTEM VENTS

SURVEILLANCE REQUIREMENT 4.4.11.2

Surveillance requirement 4.4.11.2 should be renumbered to be 4.4.11.1.C. This change would make it consistent with the standard technical specification. As presently worded, surveillance requirement 4.4.11.2 has no assigned test interval.

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REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 Two Reactor Coolant System Vent (RCSV) paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one RCSV path OPERABLE, STARTUP and/or POWER OPERATION may continue provided the inoperable path is maintained closed with power removed from the valve actuators; restore the inoperable path to OPERABLE status within 30 days; or be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

b. With no RCSV path OPERABLE, restore at least one path to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each RCSV path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the upstream manual isolation valve is locked in the opened position, and

b. Operating each remotely controlled valve through at least one complete cycle of the full travel from the control room.

c. ~~4.4.11.2~~ Each RCSV path shall be demonstrated OPERABLE by verifying flow through the RCSV paths during venting.

ACCUMULATORS

Limiting Condition for Operation 3.5.1.1

Westinghouse informed TVA that correct range for contained water volume for the cold leg accumulators is between 7779 and 8206 gallons. Westinghouse informed TVA that the correct range for the pressure for the cold leg accumulators is between 335 and 385 psig. Westinghouse informed that these values are consistent with the FSAR accident analyses.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each Cold Leg Injection Accumulator System shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between ~~7617~~ 7779 and ~~8033~~ 8206 gallons,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between ~~385~~ 335 and ~~447~~ 385 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one Cold Leg Injection Accumulator System inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one Cold Leg Injection Accumulator System inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each Cold Leg Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms or by measurement of levels and pressures, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

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EMERGENCY GAS TREATMENT SYSTEM

Action Statements - Reporting Exemptions

In a letter to NRC from D. E. McCloud to E. Adensam dated 3/24/85 (L44 850324 801), TVA requested an exemption from reporting requirements associated with technical specification 3.6.1.8 under certain circumstances. This request was made under the allowances of 10 CFR 50.73.f. This request must be resolved for certification of the technical specifications.

TABLE 3.12-1

Item 4.b should be worded similar to table 3.1 section 5.b of the Offsite Dose Calculation Manual as revised by letter dated March 20, 1985.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
b. Fish and Invertebrates	One sample each of ~~three~~ a ~~important species, including both~~ commercially and recreationally important species ~~in vicinity of plant discharge~~ from ~~area.~~ Nickajack, Chickamauga, and Watts Bar Reservoirs. ~~One sample of same species in areas not influenced by plant discharge~~.	~~Sample in season, or semiannually if they are not seasonal~~. At least once per 184 days.	Gamma isotopic analysis[5] on edible portions.
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest.[10]	Gamma isotopic analyses[5] on edible portion.
	Samples of three different kinds of available broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed as outlined in Section 4.a. above.	Monthly when available.	Gamma isotopic[5] and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed as outlined in Section 4.a. above.	Monthly when available.	Gamma isotopic[5] and I-131 analysis.

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TECHNICAL SPECIFICATION 3.5.3

The footnote at the bottom of page 3/4 5-9 and S. R. 4.5.3.2 should be modified as shown to be consistent with the APPLICABILITY statement of Mode 4. Currently the footnote requires that only one charging pump be OPERABLE at RCS temperatures less than or equal to $350^{\circ}F$. This is in direct conflict with specification 3.5.2 which requires two charging pumps be OPERABLE in Mode 3 which is defined as greater than or equal to $350^{\circ}F$. Thus, at $350^{\circ}F$ the two specifications conflict.

The proposed changes would be consistent with the handling of this same situation in specification 3.1.2.3 of the final draft technical specifications.

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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 [redacted]

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

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 allowed 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 350°F~~ by verifying that the pumps are in the pull-to-lock position and the motor circuit breakers are tagged out, or the pump(s) is isolated from the RCS by a manually closed valve or by a motor-operated valve with the valve breaker tagged. Normal seal flow can be maintained at all times.

Technical Specification Pages 3/4 6-15 and B3/4 6-3

In a March 1, 1985, letter from TVA to NRC (R. H. Shell to E. Adensam) a change to Technical Specification pages 3/4 6-15 and B3/4 6-3 was submitted. These changes are based on a report provided to TVA by Posi-Seal Inernational, the vendor for the 24-inch containment purge and vent valves. These changes to the final draft of the Watts Bar Technical Specifications have to be made for TVA to certify that Technical Specification 3.6.1.9 accurately reflects the as-built facility.

The upper and lower compartment 24-inch valves

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Attachment 2

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 Each containment purge supply and/or exhaust isolation valve shall be OPERABLE. ~~The 24-inch valves at less than or equal to 70° open,~~ the 8-inch valves, and the 12-inch valve(s) may be opened for up to 1000 hours during a calendar year provided no more than one pair (one exhaust and one supply) is open at one time.

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

ACTION:

a. With the purge supply and/or exhaust isolation valve(s) open for more than 1000 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 the 24-inch containment purge supply and/or exhaust isolation valve(s) at greater than 70° open, close the valve(s) 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.~~

c. 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, 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

[redacted]

Each 24-inch containment lower compartment purge supply and/or exhaust isolation valve(s) shall be verified at least once per 31 days to be physically restricted to less than or equal to 50° open.

With the 24-inch containment lower compartment purge supply and/or exhaust isolation valve(s) at greater than 50° open, close the valve(s) 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.

EMERGENCY GAS TREATMENT SYSTEM (Continued)

of the system with the heaters operating to maintain low humidity (≤70%)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 safety 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 24-inch upper and lower compartment valves ~~at less than or equal to 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 pair open will be limited to 1000 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 activities, 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 [redacted] valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

Justification for Change to Technical Specification
Table 3.6-2

Attached is a marked-up Table 3.6-2 from the Watts Bar Technical Specification. This change will add remote manual containment isolation valves which would be required to be open with a containment isolation signal present. This change was requested by NRC in a February 6, 1985, meeting with TVA representatives to discuss Watts Bar open technical specification issues.

The valves which are being added to Table 3.6-2 are containment isolation valves in the Watts Bar sampling system. These valves would be required to be open during post accident conditions in order to take containment air samples, reactor coolant system samples, and hydrogen analyzer samples. The opening of these valves will be performed under administrative controls.

These sampling system remote manual valves receive no automatic isolation signal; therefore, no maximum isolation time should be specified in Table 3.6-2.

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

Insert →

*Not subject to Type C leakage tests.

#The provisions of Specification 3.0.4 are not applicable if the requirements of items b or c of the ACTION statement are met.

##The provisions of Specification 3.0.4 are not applicable if the requirements of items b or c of the ACTION statement are met, and at least one other path of the affected system is OPERABLE.

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Insert for Table 3.6-2

4. Manual

FCV-43-201**	Hydrogen Analyzer	N/A
FCV-43-202**	Hydrogen Analyzer	N/A
FCV-43-207**	Hydrogen Analyzer	N/A
FCV-43-208**	Hydrogen Analyzer	N/A
FCV-43-250**	Post Accident Sampling Hot Leg 1 Train A	N/A
FCV-43-251**	Post Accident Sampling Hot Leg 1 Train A	N/A
FCV-43-287**	Post Accident Sampling Cont. Intake Train A	N/A
FCV-43-288**	Post Accident Sampling Cont. Intake Train A	N/A
FCV-43-307**	Post Accident Sampling Cont. Air Return Train A	N/A
FCV-43-309**	Post Accident Sampling Hot Leg 3 Train B	N/A
FCV-43-310**	Post Accident Sampling Hot Leg 3 Train B	N/A
FCV-43-318**	Post Accident Sampling Cont. Intake Train B	N/A
FCV-43-319**	Post Accident Sampling Cont. Intake Train B	N/A
FCV-43-325**	Post Accident Sampling Cont. Air Return Train B	N/A
FCV-43-341**	Post Accident Sampling Return to Cont. Sump Train B	N/A
FCV-43-342**	Post Accident SAmpling Return To Cont. Sump Train A	N/A

** May be opened on an intermittent basis under administrative controls..

HYDROGEN MITIGATION SYSTEM

Surveillance Requirement 4.6.4.3.b

TVA identified its proposed test method for verifying ignitor temperatures in a letter from J. W. Hufham to E. Adensam dated February 14, 1985. TVA certifies the present technical specification pending NRC documenting acceptance of the TVA test method in a supplement to the safety evaluation report. If the test method approval is not documented in a supplement, the technical specifications must be revised as requested in the letter referenced above.

Surveillance Requirement 4.6.5.1.b.3

The requirement of this section as stated in the first sentence is to verify that the thickness of frost and ice is less than or equal to 0.38 inch; however, the second sentence in this section requires remedial action if the accumulated frost or ice thickness is greater than or equal to 0.38 inch. Since a thickness of 0.38 inch is an acceptable condition, no remedial action should be required.

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one basket each from Radial Rows 1, 2, 4, 6, 8, and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1399 pounds of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1399 pounds/basket at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - Bays 1 through 8, Group 2 - Bays 9 through 16, and Group 3 - Bays 17 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8, and 9 in each group shall not be less than 1399 pounds/basket at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,719,500 pounds; and

3) Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on flow passages between ice baskets, past lattice frames, through the intermediate and top deck floor grating, or past the lower inlet plenum support structures and turning vanes is restricted to a thickness of less than or equal to 0.38 inch. If one flow passage per bay is found to have an accumulation of frost or ice with a thickness of greater than ~~or equal to~~ 0.38 inch, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.

c. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each one-third of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion or other damage. The ice baskets shall be raised at least 10 feet for this inspection.

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION 3.7.11.1.A AND SURVEILLANCE REQUIREMENT 4.7.11.1.E.2

FSAR section 9.5.1.2.1 lists the rated capcity of the fire pumps as 1590 gpm at 330 feet head. The draft technical specifications incorrectly listed the manufacturer's rated values. A review of the hydraulic calculations confirmed that a 10 percent degradation margin was assumed. End of life corrosion factors and worst-case raw service water requirements were also assumed. The correct technical specification value should be 1590 gpm at 300 feet head. This is the minimum acceptable performance value based on the hydraulic calculations. The recent preoperational test data shows some degradation below the manufacturer's pump curve; however, performance still exceeded the minimum value assumed in the hydraulic calculation.

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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 ~~three~~ fire suppression pumps, each with a capacity of 1590 gpm at ~~330~~ 300 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 one pump inoperable, restore at least three 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 testable 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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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 total pump head of ~~330~~ 300 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 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.

Table 3.7-3

Attached is a marked up copy of Table 3.7-3. These changes will make the fire hose stations in Table 3.7-3 agree with the as-built condition at Watts Bar. Also, several fire hose stations should be added. The additional stations have a unit 2 designation, but are required to protect unit 1 safety-related equipment.

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TABLE 3.7-3

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK #
Diesel Generator Building		
Corridor	742	0-26-1077
Air Exhaust 2B Room	760	0-26-1082
Entrance to 1A Elec. Bd Rm	760	0-26-1080
Reactor Building		
Reactor Coolant Pumps	702	1-26-1220
Reactor Coolant Pumps	702	1-26-1221
Reactor Coolant Pumps	702	1-26-1222
Reactor Coolant Pumps	702	1-26-1223
Reactor Coolant Pumps	702	1-26-1224
Reactor Coolant Pumps	702	1-26-1225
Standpipe R. Bldg. Annulus	Platform 702	1-26-1216
Standpipe R. Bldg. Annulus	Platform 702	1-26-1217
Standpipe R. Bldg. Annulus	Platform 702	1-26-1218
Standpipe R. Bldg. Annulus	Platform 702	1-26-1219
Standpipe R. Bldg. Annulus	Platform 724	1-26-1212
Standpipe R. Bldg. Annulus	Platform 724	1-26-1213
Standpipe R. Bldg. Annulus	Platform 724	1-26-1214
Standpipe R. Bldg. Annulus	Platform 724	1-26-1215
Standpipe R. Bldg. Annulus	Platform 744	1-26-1208
Standpipe R. Bldg. Annulus	Platform 744	1-26-1209
Standpipe R. Bldg. Annulus	Platform 744	1-26-1210
Standpipe R. Bldg. Annulus	Platform 744	1-26-1211
Standpipe R. Bldg. Annulus	Platform 763	1-26-1204
Standpipe R. Bldg. Annulus	Platform 763	1-26-1205
Standpipe R. Bldg. Annulus	Platform 763	1-26-1206
Standpipe R. Bldg. Annulus	Platform 763	1-26-1207
Standpipe R. Bldg. Annulus	Platform 782	1-26-1200
Standpipe R. Bldg. Annulus	Platform 782	1-26-1201
Standpipe R. Bldg. Annulus	Platform 782	1-26-1202
Standpipe R. Bldg. Annulus	Platform 782	1-26-1203
Standpipe R. Bldg. Annulus	Platform 801	1-26-1196
Standpipe R. Bldg. Annulus	Platform 801	1-26-1197
Standpipe R. Bldg. Annulus	Platform 801	1-26-1198
Standpipe R. Bldg. Annulus	Platform 801	1-26-1199

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TABLE 3.7-3 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK #
Auxiliary Building		
A9V	676	0-26-691
A8T	676	0-26-663
A3T	692	1-26-668
A13S	692	2-26-668
A7W	692	0-26-680
A8X	692	0-26-681
A8T	692	0-26-662
A3T	713	1-26-667
A13T	713	2-26-667
A8W	713	0-26-690
A8T	713	0-26-661
A1V	716	ABH-5, valves 1-26-674 and 1-26-675
A8X	729	0-26-658
A8X	729	0-26-659
A5X	729	1-26-686
A11X	729	2-26-686
A11Y	730	0-26-854
A3T	737	1-26-666
A8~~X~~W	737	0-26-677
A8T	737	0-26-660
A13T	737	2-26-666
A11Y	750	0-26-655
A3T	757	1-26-655
A13T	757	2-26-665
A4U	757	1-26-670
A12V	757	2-26-670
A5X	757	0-26-682
A10T	757	0-26-684
A5U	757	ABH-3, valves 1-26-671 and 1-26-672
A5X	763	1-26-693
A11X	763.5	2-26-696
A3T	772	1-26-664
A13T	772	2-26-664
A5X	775	1-26-694
A4U	782	1-26-669
A5X	~~786~~ 786.5	1-26-695
~~A5W~~	~~757~~	~~1-26-671~~
~~A5W~~	~~757~~	~~1-26-672~~
Control Building		
Stairwell C-1	692	0-26-1194
Stairwell C-1	708	0-26-1193
Stairwell C-1	729	0-26-1192
Stairwell C-1	755	0-26-1191
Stairwell C-2	692	0-26-1189
Stairwell C-2	708	0-26-1188
Stairwell C-2	729	0-26-1187
Stairwell C-2	755	0-26-1186
Intake Pumping Station (ERCW)		
Electrical Board Rm	716	0-26-595
Electrical Board Rm	716	0-26-596
B Strainer Room	727	0-26-594
A Strainer Room	727	0-26-597
A Fire Pump Room	~~727~~ 747	0-26-1710
B Fire Pump Room	~~727~~ 747	0-26-1711

TABLE 3.7-4

Items 1 and 2 -	The elevation specified for the area temperature monitors next to the 480 V shutdown board tranformers should be 772 instead of 722
Item 18 -	The temperature limits for the unit 1 additional equipment building should be greater than or equal to 70 less than or equal to 92. This will make the technical specification limits agree with the environmental data for the equipment in that building. Attached is TVA drawing 47E235-10 which shows the environmental data for the additional equipment buildings.

TABLE 3.7-4

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Aux Bldg el ~~722~~ 772 next to 480V Sd Bd transformer 1A2-A.	≤ 104
2.	Aux Bldg el ~~722~~ 772 next to 480V Sd Bd transformer 1B1-B.	≤ 104
3.	Aux Bldg el 772 next to 480V Rx MOV Bd 1A2-A.	≤ 104
4.	Aux Bldg el 772 across from spare 125V vital battery charger 1-S.	≤ 104
5.	Aux Bldg el 772 next to 480V Rx MOV Bd 2A2-A.	≤ 104
6.	Aux Bldg el 772 next to 480V Sd Bd transformer 2A2-A.	≤ 104
7.	Aux Bldg el 772 next to 480V Sd Bd transformer 2B2-B.	≤ 104
8.	Aux Bldg el 772 next to 480V Rx MOV Bd 2B2-B.	≤ 104
9.	Aux Bldg el 772 U1 Mech Equip Room B.	≤ 104
10.	Sd Bd room el 757 U1 behind stairs S-A3.	≤ 104
11.	Sd Bd room el 757 U2 behind stairs S-A13.	≤ 104
12.	Refueling floor el 757 U1 beside Aux boration makeup tk.	≤ 104
13.	Aux Bldg el 737 U1 outside supply fan room.	≤ 104
14.	Aux Bldg el 713 U1 across from AFW pumps.	≤ 104
15.	Aux Bldg el 692 U1 outside AFW pump room door.	≤ 104
16.	Aux Bldg el 692 U2 near boric acid concentrate filter vault.	≤ 104
17.	Aux Bldg el 676 next to 0-L-629.	≤ 104
18.	Add Equip Bldg U1 el 729 between UHI accumulators.	≥ ~~75~~ 70 ≤ ~~85~~ 92
19.	Main Control Room south wall.	≤ 104
20.	Main Control Room across from 1-M-9.	≤ 104
21.	D/G Bldg el 742 2B-B D/G room on wall by battery charger.	≤ 120
22.	D/G Bldg el 760.5 next to 480V diesel Aux Bd 2B1-B.	≤ 120
23.	IPS el 741 next to 1A-A ERCW-MCC transformer and board.	≤ 120
24.	IPS el 741 in B train ERCW pump room.	≤ 120
25.	IPS el 741 next to 2A-A ERCW-MCC transformer and board.	≤ 120
26.	Computer room el 708 center of room.	≥ 65 ≤ 75
27.	North steam valve vault room U1 Morgan Temp Recorder.	≥ 80
28.	South steam valve vault room U1 Morgan Temp Recorder.	≥ 80

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DOCUMENT PAGE PULLED

* OVERSIZE DUPLICATE DRAWINGS

SEE APERTURE CARDS

APERTURE CARD NO# 8404050500

AVAILABILITY ✓ PDR ___ CF ___ NMSS ___

NUMBER OF PAGES. 1

ADDITIONAL APERTURE CARD NUMBERS BELOW.

AREA TEMPERATURE MONITORING

Table 3.7-4

TVA requested a change to the temperature limits for the diesel generator rooms in a letter from J. A. Domer to E. Adensam dated March 18, 1985. This item must be resolved for certification.

TABLE 3.7-4

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Aux Bldg el 722 next to 480V Sd Bd transformer 1A2-A.	$\leq$ 104
2.	Aux Bldg el 722 next to 480V Sd Bd transformer 1B1-B.	$\leq$ 104
3.	Aux Bldg el 772 next to 480V Rx MOV Bd 1A2-A.	$\leq$ 104
4.	Aux Bldg el 772 across from spare 125V vital battery charger 1-S.	$\leq$ 104
5.	Aux Bldg el 772 next to 480V Rx MOV Bd 2A2-A.	$\leq$ 104
6.	Aux Bldg el 772 next to 480V Sd Bd transformer 2A2-A.	$\leq$ 104
7.	Aux Bldg el 772 next to 480V Sd Bd transformer 2B2-B.	$\leq$ 104
8.	Aux Bldg el 772 next to 480V Rx MOV Bd 2B2-B.	$\leq$ 104
9.	Aux Bldg el 772 U1 Mech Equip Room B.	$\leq$ 104
10.	Sd Bd room el 757 U1 behind stairs S-A3.	$\leq$ 104
11.	Sd Bd room el 757 U2 behind stairs S-A13.	$\leq$ 104
12.	Refueling floor el 757 U1 beside Aux boration makeup tk.	$\leq$ 104
13.	Aux Bldg el 737 U1 outside supply fan room.	$\leq$ 104
14.	Aux Bldg el 713 U1 across from AFW pumps.	$\leq$ 104
15.	Aux Bldg el 692 U1 outside AFW pump room door.	$\leq$ 104
16.	Aux Bldg el 692 U2 near boric acid concentrate filter vault.	$\leq$ 104
17.	Aux Bldg el 676 next to 0-L-629.	$\leq$ 104
18.	Add Equip Bldg U1 el 729 between UHI accumulators.	$\geq$ 75 $\leq$ 85
19.	Main Control Room south wall.	$\leq$ 104
20.	Main Control Room across from 1-M-9.	$\leq$ 104
21.	D/G Bldg el 742 2B-B D/G room on wall by battery charger.	$\leq$ 120
22.	D/G Bldg el 760.5 next to 480V diesel Aux Bd 2B1-B.	$\leq$ 120
23.	IPS el 741 next to 1A-A ERCW-MCC transformer and board.	$\leq$ 120
24.	IPS el 741 in B train ERCW pump room.	$\leq$ 120
25.	IPS el 741 next to 2A-A ERCW-MCC transformer and board.	$\leq$ 120
26.	Computer room el 708 center of room.	$\geq$ 65 $\leq$ 75
27.	North steam valve vault room U1 Morgan Temp Recorder.	$\geq$ 80
28.	South steam valve vault room U1 Morgan Temp Recorder.	$\geq$ 80
29	D/G BLDG el 742 1A-A D/G Room Near D/G SET	$\geq$ 40
30	D/G BLDG el 742 1B-B D/G Room Near D/G SET	$\geq$ 40
31	D/G BLDG el 742 2A-A D/G Room Near D/G SET	$\geq$ 40
32	D/G BLDG el 742 2B-B D/G ROOM Near D/G SET	$\geq$ 40

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Surveillance Requirements 4.8.1.1.2.c and 4.8.1.1.2.e

In a meeting with NRC Power Systems Branch on February 28, 1985, TVA was requested to do additional fuel oil sampling because of the Watts Bar design of the 7-day fuel oil tanks. The technical specification changes associated with the additional sampling will be submitted in a separate letter but will need to be resolved for certification of the technical specifications.

ELECTRICAL POWER SYSTEMS

Surveillance Requirement 4.8.1.1.2.h.2

TVA requested a change to surveillance requirement 4.8.1.1.2.h.2 in a letter from J.A. Domer to E. Adensam dated March 18, 1985. This item must be resolved for certification.

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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 do not exceed the 2000-hour rating of 4400 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 automatic load sequence timers are OPERABLE and their Setpoints are within the specified bands; and

11) 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.

12)

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 900 ± 18 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 repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 4400 kW for 1 hour or until operating temperature has stabilized.

12) Performing a visual inspection for leaks in the exposed fuel oil piping while the diesel generator is running.

SURVEILLANCE REQUIREMENTS (Continued)

h. 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 125 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 100 volts or a battery overcharge with a battery terminal voltage above 135 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 150×10^{-6} ohm, 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 anticorrosion material, and

Technical Specification Table 3.8-1

Attached are marked-up pages from Technical Specification Table 3.8-1. Also attached are pages showing additional containment penetration conductor overcurrent protective devices which should be included in Table 3.8-1. Some additions of these result from a decision to include all 125 volt dc and 120 volt ac fuses in this table pending a reverification of the calculations to justify exempting them. Exemption was previously justified because calculations showed that insufficient energy was available in the circuits to damage the penetrations. The other additions result from an oversight in previous submittals.

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
1. 6.9K RCP Boards			
52-202 -2/1A	52-2112	6.9kV RCP BD 1A	REAC Coolant Pump 1
52-202 -2/1B	52-2114	6.9kV RCP BD 1B	REAC Coolant Pump 2
52-202 -2/1C	52-2122	6.9kV RCP BD 1C	REAC Coolant Pump 3
52-202 -3/1D	52-2124	6.9kV RCP BD 1~~B~~D	REAC Coolant Pump 4
2. 480V Boards			
52-212 -7B/A1	FU-212 -A17/13	Shutdown BD 1A1-A	CRD MECH CLR FAN 1A-A/1
52-212 -7D/A1	FU-212 -A17/~~13~~33	Shutdown BD 1A1-A	CRD MECH CLR FAN 1A-A/2
52-212 -7C/A1	FU-212 -A17/23	Shutdown BD 1A1-A	REAC LWR COMP CLR FAN 1A-A
52-212 -10C/A1	FU-212 -A110/23	Shutdown BD 1A1-A	CNTMT AIR RTN FAN 1A-A
52-212 -7A/A2	FU-212 -A27/3	Shutdown BD 1A2-A	CRD MECH CLR FAN 1C-A ~~Supply~~
52-212 -7D/A2	FU-212 -A27/33	Shutdown BD 1A2-A	REAC LWR COMPT CLR FAN 1C-A
52-212 -8A/A2	FU-212 -A28/3	Shutdown BD 1A2-A	CRD MECH CLR FAN 1C-A/2
52-212 -7C/B1	FU-212 -B17/23	Shutdown BD 1B1-B	CRD MECH CLR FAN 1B-B/1

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE	SYSTEM POWERED
2. 480V Boards (Continued)			
52-212 -7D/B1	FU-212 -B17/33	Shutdown BD 1B1-B	REAC LWR COMP CLR FAN 1B-B
52-212 -~~7D~~/B1 10C	FU-212 -B110/~~23~~ 23	Shutdown BD 1B1-B	CRD MECH CLR FAN 1B-B/2
52-212 -3B/B2	FU-212 -B23/12	Shutdown BD 1B2-B	REAC BLDG CRANE
52-212 -7B/B2	FU-212 -B27/13	Shutdown BD 1B2-B	CRD MECH CLR FAN 1D-B/1
52-212 -7D/B2	FU-212 -B27/33	Shutdown BD 1B2-B	REAC LWR CDM CLR FAN 1D-B
52-212 -9C/B2	FU-212 -B29/~~23~~ 23	Shutdown BD 1B2-B	CNTMNT AIR RTN FAN 1B-B
52-212 -10C/B2	FU-212 -B210/23	Shutdown BD 1B2-B	CRD MECH CLR FAN 1D-B/2
52-213 -7D/A1	FU-213 -A17/32	REAC MOV BD 1A1-A	SIS ACC TK 3 FLOW ISLN VLV
52-213 -8D/A1	FU-213 -A18/32	REAC MOV BD 1A1-A	SIS ACC TK 1 FLOW ISLN VLV
52-213 -5B/A1	FU-213 -A110/32	REAC MOV BD 1A1-A	RHR SYS ISLN VLV
52-213 -6D/A1	FU-213 -A16/32	REAC MOV BD 1A1-A	RCS PRESS RELIEF FLOW CONT VLV
52-213 -16A/A1	FU-213 -A116/2	REAC MOV BD 1A1-A	INCORE INSTR RM CLR FAN 1A
52-213 -16B/A1	FU-213 -A116/12	REAC MOV BD 1A1-A	REACT CNTMT PIT SMP PMP ETCTR
52-213 -17E/A1	FU-213 -A117/42	REAC MOV BD 1A1-A	POWER OUTLETS

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE	SYSTEM POWERED
2. 480V Boards (Continued)			
52-213 -6D/A2	FU-213 -A26/32	REAC MOV BD 1A2-A	RHR SYS ISLN BYPASS VLV
52-213 -7D/A2	FU-213 -A27/32	REAC MOV BD 1A2-A	LWR CNTMT 1A CLR DISCH ISLN VLV
52-213 -8D/A2	FU-213 -A28/32	REAC MOV BD 1A2-A	LWR CNTMT 1C CLRS DISCH ISLN VLV
52-213 -9D/A2	FU-213 -A29/32	REAC MOV BD 1A2-A	UPR CNTMT VT CLR 1A DISCH ISLN VLV
Insert A			
52-213 -6D/B2	FU-213 -B212/32	REAC MOV BD 1B2-B	RCP THRM BAR RTN CNTMT ISLN VLV
52-213 -7D/B2	FU-213 -B27/32	REAC MOV BD 1B2-B	LWR CNTMT 1B CLRS DISCH ISLN VLV
52-213 -8D/B2	FU-213 -B28/32	REAC MOV BD 1B2-B	LWR CNTMT 1D CLRS DISCH ISLN VLV
52-213 -9D/B2	FU-213 -B29/32	REAC MOV BD 1B2-B	UPR CNTMT VT CLR 1B DISCH ISLN VLV
52-213 -10D/B2	FU-213 -B210/32	REAC MOV BD 1B2-B	UPR CNTMT VT CLR 1D DISCH ISLN VLV
52-213 -13D/B2	FU-213 -B213/32	REAC MOV BD 1B2-B	RCP OIL CLR RTN CNTMT ISLN VLV
52-232 -2A/1A	FU-232 -A2/2	REAC VENT BD 1A-A	CNTMT FL & EQ DR SMP PMP 1A
52-232 -3A/1A	FU-232 -A3/2	REACT VENT BD 1A-A	INCORE FLUX DET DRIVE UNIT 1D
52-232 -3B/1A	FU-232 -A3/12	REAC VENT BD 1A-A	INCORE FLUX DET DRIVE UNIT 1E
52-232 -3C/1A	FU-232 -A3/22	REAC VENT BD 1A-A	INCORE FLUX DET DRIVE UNIT 1F

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY ~~DEVICES~~ DEVICE	SYSTEM POWERED
2. 480V Boards (Continued)			
52-232 -7B/1A	FU-232 -A7/12	REAC VENT BD 1A-A	RCP 1 MTR HTR
52-232 -7D/1A	FU-232 -A7/32	REAC VENT BD 1A-A	RCP 1 OIL LIFT PMP
52-232 -8B/1A	FU-232 -A8/12	REAC VENT BD 1A-A	RCP 3 MTR HTR
52-232 -8D/1A	FU-232 -A8/32	REAC VENT BD 1A-A	RCP 3 OIL LIFT PMP
52-232 -9B/1A	FU-232 -A9/12	REAC VENT BD 1A-A	REC UPR CCMPT CLR FAN 1A
52-232 10B/1A	FU-232 -A10/12	REAC VENT BD 1A-A	REC UPR CMPNT CLR FAN 1C
52-232 -10F/1A	FU-232 -A10/52	REAC VENT BD 1A-A	STUD TENSION HOISTS
52-232 -11D/1A	FU-232 -A11/32	REAC VENT BD 1A-A	REAC COOL. DRAIN TK PMP 1A
52-232 11F/1A	FU-232 -A11/52	REAC VENT BD 1A-A	REAC LWR COMPT U HTR 1A
52-232 -12F/1A	FU-232 -A12/52	REAC VENT BD 1A-A	CNTMT INSTR RM U HTR 1A
52-232 -13A/1A	FU-232 -A13/2	REAC VENT BD 1A-A	REAC BLDG MANI CRN 1
52-232 -13D/1A	FU-232 -A13/~~2~~ 32	REAC VENT BD 1A-A	IC AHU (S)
52-232 -14B/1A	FU-232 -A14/12	REAC VENT BD 1A-A	IC END WALL DOOR 1B
52-232 -14D/1A	FU-232 -A14/32	REAC VENT BD 1A-A	IC AHU (S)

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE	SYSTEM POWERED
2. 480V Boards (Continued)			
52-232 -15A/1A	FU-232 -A15/2	REAC VENT BD 1A-A	REAC UPR COMPT HTR 1A
52-232 -16A/1A	FU-232 -A16/2	REAC VENT BD 1A-A	REAC UPR COMP HTR 1C
52-232 -2D/1A	FU-232 -A2/32	REAC VENT BD 1A-A	H2 ELECT RECOMBINER 1A-A
52-232 -2D/1B	FU-232 -B2/32	REAC VENT BD 1B-B	H2 ELECT RECOMBINER 1B-B
52-232 -2A/1B	FU-232 -B2/2	REAC VENT BD 1B-B	CNTMT FL & EQPT DRAIN SMP PMP 1B
52-232 -3A/1B	FU-232 -B3/2	REAC VENT BD 1B-B	INCORE FLUX DET DRIVE UNIT 1A
52-232 -3B/1B	FU-232 -B3/12	REAC VENT BD 1B-B	INCORE FLUX DET DRIVE UNIT 1B
52-232 -3C/1B	FU-232 -B3/22	REAC VENT BD 1B-B	INCORE FLUX DET DRIVE UNIT 1C
52-232 -7B/1B	FU-232 -B7/~~22~~ 12	REAC VENT BD 1B-B	RCP 2 MTR HTR
52-232 -7D/1B	FU-232 -B7/32	REAC VENT BD 1B-B	RCP 2 OIL LIFT PUMP
52-232 -8B/1B	FU-232 -B8/12	REAC VENT BD 1B-B	RCP 4 MTR HTR
52-232 -8D/1B	FU-232 -B8/32	REAC VENT BD 1B-B	RCP 4 OIL LIFT PUMP
52-232 -9B/1B	FU-232 -B9/12	REAC VENT BD 1B-B	REAC UPR COMPT CLR FAN 1B
52-232 -10B/1B	FU-232 -B10/12	REAC VENT BD 1B-B	REAC UPR COMPT CLR FAN 1D

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
2. 480V Boards (Continued)			
52-232 -10F/1B	FU-232 -B10/52	REAC VENT BD 1B-B	REAC BLDG JIB CRN
52-232 -11D/1B	FU-232 -B11/32	REAC VENT BD 1B-B	REAC COOL DR TK PMP 1B
52-232 -11F/1B	FU-232 -B11/52	REAC VENT BD 1B-B	REAC LWR COMPT U HTR 1B
52-232 -12F/1B	FU-232 -B12/52	REAC VENT BD 1B-B	CNTMT INST RM U HTR 1B
52-232 -13B/1B	FU-232 -B13/~~[illegible]~~ 12	REAC VENT BD 1B-B	IC END WALL DR 1B
52-232 -13D/1B	FU-232 -B13/32	REAC VENT BD 1B-B	IC AHU(S)
52-232 -13F/1B	FU-232 -B13/52	REAC VENT BD 1B-B	IC BRIDGE CRN
52-232 -14B/1B	FU-232 -B14/12	REAC VENT BD 1B-B	RCC CHANGE HOIST
52-232 -14D/1B	FU-232 -B14/~~[illegible]~~ 32	REAC VENT BD 1B-B	IC AHU(S)
52-232 -14F/1B	FU-232 -B14/52	REAC VENT BD 1B-B	EQPT HATCH HOIST
52-232 -15A/1B	FU-232 -B15/2	REAC VENT BD 1B-B	REAC UPR COMPT HTR 1B
52-232 -16A/1B	FU-232 -B16/2	REAC VENT BD 1B-B	REAC UPR COMPT HTR 1D
3. 480V AC CAB			
CB-68 -341F/D1	FU-211 -A21/X10	DIST CAB CONT GP 1D	PRESSURIZER HTRS GP 1D ELEMENTS 47, 49, 51

Insert B

TABLE 3.8-1 (Continued) FINAL DRAFT

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
3. 480V AC CAB (Continued)			
CB-68 -341F/D2	FU-211 -A21/~~X~~ 11	DIST CAB CONT GP 1D	PRESSURIZER HTRS GP 1D ELEMENTS 52, 54, 56
CB-68 -341F/D3	FU-211 -A21/~~X~~ 12	DIST CAB CONT GP 1D	PRESSURIZER HTRS GP 1D ELEMENTS 57, 59, 61
CB-68 -341F/D4	FU-211 -A21/~~X~~ 15	DIST CAB CONT GP 1D	PRESSURIZER HTRS GP 1D ELEMENTS 72, 74, 76
CB-68 -341F/D5	FU-211 -A21/~~X~~ 14	DIST CAB CONT GP 1D	PRESSURIZER HTRS GP 1D ELEMENTS 67, 69, 71
CB-68 -341F/D6	FU-211 -A21/~~X~~ 13	DIST CAB CONT GP 1D	PRESSURIZER HTRS GP 1D ELEMENTS 66, 64, 62
CB-68 -341A/A1-A	FU-211 -A20/~~X~~ 11	DIST CAB CONT GP 1A-A	PRESSURIZER HTRS GP 1A-A ELEMENTS 22, 24, 26
CB-68 -341A/A2-A	FU-211 -A20/~~X~~ 12	DIST CAB CONT GP 1A-A	PRESSURIZER HTRS GP 1A-A ELEMENTS 28, 30, 32
CB-68 -341A/A3-A	FU-211 -A20/~~X~~ 13	DIST CAB CONT GP 1A-A	PRESSURIZER HTRS GP 1A-A ELEMENTS 34, 36, 38

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE	SYSTEM POWERED
3. 480V AC CAB (Continued)			
CB-68 -341A/A7-A	FU-211 -A20/17	DIST CAB CONT GP 1A-A	PRESSURIZER HTRS GP 1A-A ELEMENTS 17, 15, 19
CB-68 -341A/A6-A	FU-211 -A20/16	DIST CAB CONT GP 1A-A	PRESSURIZER HTRS GP 1A-A ELEMENTS 11, 9, 13
CB-68 -341A/A7-A	FU-211 -A20/15	DIST CAB CONT GP 1A-A	PRESSURIZER HTRS GP 1A-A ELEMENTS 5, 3, 7
CB-68 -341A/A8-A	FU-211 -A20/14	DIST CAB CONT GP 1A-A	PRESSURIZER HTRS GP 1A-A ELEMENTS 42, 40, 44
CB-68 -341D/B1-B	FU-211 -B20/11	DIST CAB CONT GP 1B-B	PRESSURIZER HTRS GP 1B-B ELEMENTS 23, 25, 27
CB-68 -341D/B2-B	FU-211 -B20/12	DIST CAB CONT GP 1B-B	PRESSURIZER HTRS GP 1B-B ELEMENTS 29, 31, 33
CB-68 -341D/B3-B	FU-211 -B20/13	DIST CAB CONT GP 1B-B	PRESSURIZER HTRS GP 1B-B ELEMENTS 35, 37, 39
CB-68 -341D/B5-B	FU-211 -B20/17	DIST CAB CONT GP 1B-B	PRESSURIZER HTRS GP 1B-B ELEMENTS 16, 14, 18
CB-68 -341D/B6-B	FU-211 -B20/16	DIST CAB CONT GP 1B-B	PRESSURIZER HTRS GP 1B-B ELEMENTS 10, 8, 12

TABLE 3.8-1 (Continued) **FINAL DRAFT**

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE	SYSTEM POWERED
3. 480V AC CAB (Continued)			
CB-68 -341D/B7-B	FU-211 -B20/~~X~~-15	DIST CAB CONT GP 1B-B	PRESSURIZER HTRS GP 1B-B ELEMENTS 4, 2, 6
CB-68 -341D/B8-B	FU-211 -B20/~~X~~-14	DIST CAB CONT GP 1B-B	PRESSURIZER HTRS GP 1B-B ELEMENTS 43, 41, 45
CB-68 -341H/C1	FU-211 -B21/~~X~~-10	DIST CAB CONT GP 1C	PRESSURIZER HTRS GP 1C ELEMENTS 1, 21, 48
CB-68 -341H/C2	FU-211 -B21/~~X~~-11	DIST CAB CONT GP 1C	PRESSURIZER HTRS GP 1C ELEMENTS 50, 53, 55
CB-68 -341H/C3	FU-211 -B21/~~X~~-12	DIST CAB CONT GP 1C	PRESSURIZER HTRS GP 1C ELEMENTS 58, 60, 63
CB-68 -341H/C4	FU-211 -B21/~~X~~-13	DIST CAB CONT GP 1C	PRESSURIZER HTRS GP 1C ELEMENTS 20, 46, 78
CB-68 -341H/C5	FU-211 -B21/~~X~~-14	DIST CAB CONT GP 1C	PRESSURIZER HTRS GP 1C ELEMENTS 73, 75, 77
CB-68 -341H/C6	FU-211 -B21/~~X~~-15	DIST CAB CONT GP 1C	PRESSURIZER HTRS GP 1C ELEMENTS 65, 68, 70

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

	PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICES	SYSTEM POWERED
	4. MISC 125V DC Control Pwr			
Insert C	FU-212 -A17/11N(+)	~~BKR-202 -BDI/PNL 2~~ FU-212 -A17/11N(-)	480V SHUT DN BD 1A1-A	CRD MECH CLR FAN 1A-A/1
	~~FU-212 -A17/31~~	~~BKR-202 -BDI/PNL 2~~	~~480V SHUT DN BD 1A1-A~~	~~CRD MECH CLR FAN 1A-A/2~~
Insert D	FU-212 -A17/21N(+)	~~BKR-202 -BDI/PNL 2~~ FU-212 -A17/21N(-)	480V SHUT DN BD 1A1-A	REAC LWR COMPT CLR FAN 1A-A
Insert E	FU-212 -A110/21(+)	~~BKR-202 -BDI/PNL 2~~ FU-212 -A110/21(-)	480V SHUT DN BD 1A1-A	CNTMNT AIR RTN FAN 1A-A
Insert F	FU-212 -A27/1N(+)	~~BKR-203 -BDI/PNL 2~~ FU-212 -A27/1N(-)	480V SHUT DN BD 1A2-A	CRD MECH CLR FAN 1C-A/1
Insert G	FU-212 -A27/31N(+)	~~BKR-203 -BDI/PNL 2~~ FU-212 -A27/31N(-)	480V SHUT DN DB 1A2-A	REAC LWR COMPT CLR FAN 1C-A
	~~FU-212 -A28/1~~	~~BKR-203 -BDI/PNL 2~~	~~480V SHUT DN BD 1A2-A~~	~~CRD MECH CLR FAN 1C-A/2~~
	~~FU-212 -B17/21N~~	~~BKR-202 -BDII/PNL 2~~	~~480V SHUT DN BD 1B1-B~~	~~REAC LWR COMPT CLR FAN 1C-A/2~~
Insert H	FU-212 -B17/31N(+)	~~BKR-202 -BDII/PNL 2~~ FU-212 -B17/31N(-)	480V SHUT DN BD 1B1-B	REAC LWR COMP CLR FAN 1B-B
Insert I	FU-212 -~~B120~~ 17/21N(+)	~~BKR-202 -BDII/PNL 2~~ FU-212 -B17/21N(-)	480V SHUT DN BD 1B1-B	CRD MECH CLR FAN 1B-B/~~2~~1
	~~FU-212 -A110/22~~	~~BKR-202 -BDII/PNL 2~~	~~480V SHUT DN BD 1A1-A~~	~~CNTMT AIR RTN FAN 1A-A~~
	FU-212 -B23/11(+)	~~BKR-206 -BDIV/PNL 2~~ FU-212 -B23/11(-)	480V SHUT DN BD 1B2-B	REAC BLDG ~~[illegible]~~ CRANE
Insert J	FU-212 -B27/11N(+)	~~BKR-203 -BDII/PNL 2~~ FU-212 -B27/11N(-)	480V SHUT DN BD 1B2-B	CRD MECH CLR FAN 1D-B/1

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
4. MISC 125V DC Control Pwr (Continued)			
FU-212 -B27/31N(+)	FU-212 -B27/31N(-) ~~BKR-203 BDII/PNL 2~~	480V SHUT DN BD 1B2-B	REAC LWR COMP CLR FAN 1D-B
Insert K			
FU-212 -B29/21(+)	FU-212 -B29/21(-) ~~BKR-203 BDII/PNL 2~~	480V SHUT DN BD 1B2-B	CNTMT AIR RTN FAN 1B-B
Insert L			
~~FU-212 -B210/21~~	~~BKR-203 BDII/PNL 2~~	~~480V SHUT DN BD 1B2-B~~	~~CRD MECH CLR FAN 1D-B/2~~
~~FU-212 -B29/22~~	~~BKR-203 BDII/PNL 2~~	~~480V SHUT DN BD 1B2-B~~	~~CNTMT AIR RTN FAN 1B-B~~
5. 125V DC (VI-PWR)			
FU-236 -1/A1(+)	~~52-236 310/I~~ FU-236 -1/A1(-)	125V VI-BATT BDI PNL 4	CHGRG FLOW TO RCS COOL LP 4
FU-236 -1/A3(+)	~~52-236 310/I~~ FU-236 -1/A3(-)	125V VI-BATT BDI PNL 4	RCP LP 3 LT DN FLOW ~~[illegible]~~
Insert M			
FU-236 -1/A6(+)	~~52-236 310/I~~ FU-236 -1/A6(-)	125V VI-BATT BDI PNL 4	REGEN HT EXCH LET DN ISLN VLV A
FU-236 -1/A7(+)	~~52-236 310/I~~ FU-236 -1/A7(-)	125V VI-BATT BDI PNL 4	REGEN HT EXCH ~~RCP LP 3 LT DN FLOW VLV [illegible]~~ LET DN ISLN VLV B
FU-236 -1/A8(+)	~~52-236 310/I~~ - 1/A8(-)	125V VI-BATT BDI PNL 4	REGEN HT EXCH ~~RCP LP 3 LT DN FLOW VLV [illegible]~~ LET DN ISLN VLV C
FU-236 -1/A9(+)	~~52-236 310/I~~ FU-236 -1/A9(-)	125V VI-BATT BDI PNL 4	RCP LP 3 LT DN FLOW ~~[illegible]~~
FU-236 -1/A11(+)	~~52-236 310/I~~ FU-236 -1/A11(-)	125V VI-BATT BDI PNL 4	PRESS PWR RLF VLV ~~[illegible]~~
FU-236 -1/A17(+)	~~52-236 310/I~~ FU-236 -1/A17(-)	125V VI-BATT BDI PNL 4	REGEN HT EXCH LET DN ISLN VLV
Insert N			
FU-236 -1/A21(+)	~~52-236 310/I~~ FU-236 -1/A21(-)	125V VI-BATT BDI PNL 4	LWR CNTMT VT CLR A SPLY VLV

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY ~~DEVICES~~ DEVICE	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -1/A22 (+)	~~52-236 -31071~~ FU-236 -1/A22 (-)	125V VI-BATT BDI PNL 4	CRD VT CLR A SPLY VLV
FU-236 -1/A23 (+)	~~52-236 -31071~~ FU-236 -1/A23 (-)	125V VI-BATT BDI PNL 4	LWR CNTMT VT CLR C SPLY VLV
FU-236 -1/A24 (+)	~~52-236 -31071~~ FU-236 -1/A24 (-)	125V VI-BATT BDI PNL 4	CRD VT CLR ~~D~~ C SPLY VLV
FU-236 -1/A31 (+)	~~52-236 -31071~~ FU-236 -1/A31 (-)	125V VI-BATT BDI PNL 4	SIS CK VLV ISLN HDR FLOW ISLN VLV
FU-236 -1/A40 (+)	~~52-236 -31071~~ FU-236 -1/A40 (-)	125V VI-BATT BDI PNL 4	RCP MOT CLR A SPLY VLV
FU-236 -1/A41 (+)	~~52-236 -31071~~ FU-236 -1/A41 (-)	125V VI-BATT BDI PNL 4	RCP MOT CLR C SPLY VLV
FU-236 -1/A42 (+)	~~52-236 -31071~~ FU-236 -1/A42 (-)	125V VI-BATT BDI PNL 4	CRD CLG UNIT 1A-A SUCT DMPR
FU-236 -1/A43 (+)	~~52-236 -31071~~ FU-236 -1/A43 (-)	125V VI-BATT BDI PNL 4	CRD CLG UNIT 1A-A RM DIVR DMPR
FU-236 -1/A44 (+)	~~52-236 -31071~~ FU-236 -1/A44 (-)	125V VI-BATT BDI PNL 4	CRD CLG UNIT 1C-A SUCT DMPR
FU-236 -1/A45 (+)	~~52-236 -31071~~ FU-236 -1/A45 (-)	125V VI-BATT BDI PNL 4	CRD CLG UNIT 1C-A RM DIVR DMPR
Insert O			
FU-236 -1/B5 (+)	~~52-236 -31171~~ FU-236 -1/B5 (-)	125V VI-BATT BDI PNL 4	CHRG FLOW TO RCS SPRAY
Insert P			
FU-236 -1/B17 (+)	~~52-236 -31171~~ FU-236 -1/B17 (-)	125V VI-BATT BDI PNL 4	TEST LINE ISLN VLV ~~[illegible]~~
FU-236 -1/B18 (+)	~~52-236 -31171~~ FU-236 -1/B18 (-)	125V VI-BATT BDI PNL 4	TEST LINE ISLN VLV ~~[illegible]~~
Insert Q			

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -1/B29 (+)	~~52-236~~ FU-236 ~~311/I~~ -1/B29 (-)	125V VI-BATT BDI PNL 4	UPR COMPT PURGE ISLN VLV
FU-236 -1/B30 (+)	~~52-236~~ FU-236 ~~311/I~~ -1/B30 (-)	125V VI-BATT BDI PNL 4	LWR COMPT PURGE ISLN VLV
FU-236 -1/B32 (+)	~~52-236~~ FU-236 ~~311/I~~ -1/B32 (-)	125V VI-BATT BDI PNL 4	INCOR INSTR RM PURGE ISLN VLV
Insert S FU-236 -1/B36 (+)	~~52-236~~ FU-236 ~~311/I~~ -1/B36 (-)	125V VI-BATT BDI PNL 4	LWR COMPT PURGE VLV
FU-236 -1/C3 (+)	~~52-236~~ FU-236 ~~312/I~~ -1/C3 (-)	125V VI-BATT BDI PNL 4	INSTR RM COOL UNIT A VLV
FU-236 -1/C4 (+)	~~52-236~~ FU-236 ~~312/I~~ -1/C4 (-)	125V VI-BATT BDI PNL 4	INSTR RM COOL UNIT A VLV
FU-236 -1/C22 (+)	~~52-236~~ FU-236 ~~312/I~~ -1/C22 (-)	125V VI-BATT BDI PNL 4	ST GEN BLDN ISLN VLV LP 1
FU-236 -1/C38 (+)	~~52-236~~ FU-236 ~~312/I~~ -1/C38 (-)	125V VI-BATT BDI PNL 4	ST GEN BLDN ISLN VLV LP ~~1~~ 3
FU-236 -1/C46 (+)	~~52-236~~ FU-236 ~~312/I~~ -1/C46 (-)	125V VI-BATT BDI PNL 4	LWR COMPT PURGE ISLN VLV
~~FU-236 -1/C48~~	~~52-236 312/I~~	~~125V VI-BATT BDI PNL 4~~	~~CNTMT ANNS DIFF PRESS ISLN VLV~~
FU-236 -1/C49 (+)	~~52-236~~ FU-236 ~~312/I~~ -1/C49 (-)	125V VI-BATT BDI PNL 4	LOCA H_2 CNTMT ISLN VLV
FU-236 -1/C50 (+)	~~52-236~~ FU-236 ~~312/I~~ -1/C50 (-)	125V VI-BATT BDI PNL 4	LOCA H_2 CNTMT ISLN VLV
Insert T FU-236 -1/D2 (+)	~~52-236~~ FU-236 ~~217/I~~ -1/D2 (-)	125V VI-BATT BDI PNL 4	RCP 3 SEAL RTN FLOW CONT ~~XXX~~ VLV
Insert U FU-236 -1/D17 (+)	~~52-236~~ FU-236 ~~217/I~~ -1/D17 (-)	125V VI-BATT BDI PNL 4	RCP 1 SEAL RTN FLOW CONT ~~XXX~~ VLV

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -1/D22 (+)	~~52-236~~ FU-236 ~~21971~~ -1/D22(-)	125V VI-BATT BDI PNL 4	GLYCOL SPLY FROM EXP TK ~~FSV-[illegible]~~
Insert V			
FU-236 -1/D33(+)	~~52-236~~ FU-236 ~~21971~~ -1/D33(-)	125V VI-BATT BDI PNL 4	RCP 1 STD PIPE MAKEUP WTR VLV
Insert W			
FU-236 -1/D41(+)	~~52-236~~ FU-236 ~~21771~~ -1/D41(-)	125V VI-BATT BDI PNL 4	GLYCOL SPLY FROM ~~FROM~~ EXP TK ~~FSV-[illegible]-11~~
Insert X			
FU-236 -1/E34(+)	~~52-236~~ FU-236 ~~21871~~ -1/E34(-)	125V VI-BATT BDI PNL 4	RCP 3 STD PIPE MAKEUP WTR VLV
FU-236 -1/E36 (+)	~~52-236~~ FU-236 ~~21871~~ -1/E36(-)	125V VI-BATT BDI PNL 4	RCS LP 3 HOT LEG FEED TEST LINE VLV
FU-236 -2/A3 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A3(-)	125V VI-BATT BDII PNL 4	CHGRG FLOW RCS COOL LP 1
FU-236 -2/A4 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A4(-)	125V VI-BATT BDII PNL 4	PRESS~~R~~ RLF TK ISLN VLV
FU-236 -2/A5 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A5(-)	125V VI-BATT BDII PNL 4	PRESS~~R~~ ~~[illegible]~~ PWR RLF ~~[illegible]~~ VLV
FU-236 -2/A8 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A8(-)	125V VI-BATT BDII PNL 4	EXCESS LETDN DIVR FLOW CONT VLV
FU-236 -2/A9 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A9(-)	125V VI-BATT BDII PNL 4	LWR CNTMT VT CLR B SPLY VLV
FU-236 -2/A10 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A10(-)	125V VI-BATT BDII PNL 4	CRD VT CLR B SPLY VLV
FU-236 -2/A11 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A11(-)	125V VI-BATT BDII PNL 4	LWR CNTMT VT CLR D SPLY VLV
FU-236 -2/A12 (+)	~~52-236~~ FU-236 ~~310/II~~ -2/A12(-)	125V VI-BATT BDII PNL 4	CRD VT CLR D SPLY VLV

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -2/A13(+)	~~52-236-310711~~ FU-236 -2/A13(-)	125V VI-BATT BDII PNL 4	CRD CLG UNIT 1D-B SUCT DMPR
FU-236 -2/A14(+)	~~52-236-310711~~ FU-236 -2/A14(-)	125V VI-BATT BDII PNL 4	CRD CLG UNIT 1D-B RM DIVR DMPR
FU-236 -2/A15(+)	~~52-236-310711~~ FU-236 -2/A15(-)	125V VI-BATT BDII PNL 4	CRD CLG UNIT 1B-B SUCT DMPR
FU-236 -2/A16(+)	~~52-236-310711~~ FU-236 -2/A16(-)	125V VI-BATT BDII PNL 4	CRD CLG UNIT 1B-B RM DIVR DMPR
FU-236 -2/A17(+)	~~52-236-310711~~ FU-236 -2/A17(-)	125V VI-BATT BDII PNL 4	GLYCOL SPLY ISLN VLV ~~[illegible]~~
FU-236 -2/A18(+)	~~52-236-310711~~ FU-236 -2/A18(-)	125V VI-BATT BDII PNL 4	GLYCOL ~~SPLY~~ RTN ISLN VLV ~~[illegible]~~
FU-236 -2/A20(+)	~~52-236-310711~~ FU-236 -2/A20(-)	125V VI-BATT BDII PNL 4	PRESS GAS ISLN VLV
FU-236 -2/A21(+)	~~52-236-310711~~ FU-236 -2/A21(-)	125V VI-BATT BDII PNL 4	PRESS LIQ ISLN VLV
FU-236 -2/A22(+)	~~52-236-310711~~ FU-236 -2/A22(-)	125V VI-BATT BDII PNL 4	RCS HOT LEG LP 1 OR 3 ISLN VLV
Insert Y			
~~FU-236 -2/B32(+)~~ NOW C40	~~52-236-310711~~ ~~FU-236 -2/B32(-)~~	~~125V VI-BATT BDII PNL 4~~	~~LWR COMPT PURGE ISLN VLVS~~
FU-236 -2/B33(+) see "AA"	~~52-236-313711~~ FU-236 -2/B33(-)	125V VI-BATT BDII PNL 4	CNTMT BLDG UPR COMPT AIR MON ISLN VLV ~~[illegible]~~
~~FU-236 -2/B34(+)~~ NOW C37	~~52-236-313711~~ ~~FU-236 -2/B34(-)~~	~~125V VI-BATT BDII PNL 4~~	~~INSTR RM PURGE ISLN VLVS~~

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -2/B36 (+)	~~52-236 312711~~ FU-236 -2/B36 (-)	125V VI-BATT BDII PNL 4	CNTMT BLDG UPR COMPT AIR MON ISLN VLV ~~[illegible]~~
FU-236 -2/C5 (+)	~~52-236 312711~~ FU-236 -2/C5 (-)	125V VI-BATT BDII PNL 4	INSTR RM COOL UNIT B VLV
FU-236 -2/C6 (+)	~~52-236 312711~~ FU-236 -2/C6 (-)	125V VI-BATT BDII PNL 4	INSTR RM COOL UNIT B VLV
FU-236 -2/C7 (+)	~~52-236 312711~~ FU-236 -2/C7 (-)	125V VI-BATT BDII PNL 4	CNTMT ANNS DIFF PRESS ISLN VLV
FU-236 -2/C10 (+)	~~52-236 312711~~ FU-236 -2/C10 (-)	125V VI-BATT BDII PNL 4	LOCA H_2 CNTMT ISLN VLV
FU-236 -2/C11 (+)	~~52-236 312711~~ FU-236 -2/C11 (-)	125V VI-BATT BDII PNL 4	LOCA H_2 CNTMT ISLN VLV
FU-236 -2/C17 (+)	~~52-236 312711~~ FU-236 -2/C17 (-)	125V VI-BATT BDII PNL 4	STM GEN BLDN ISLN VLV LP 2
Insert Z			
~~FU-236 -2/C21~~	~~52-236 312711~~	~~125V VI-BATT BDII PNL 4~~	~~DRSTR EXCESS LTDN HX TO HOT SMPLG RM ISLN VLV~~
FU-236 -2/C24 (+)	~~52-236 312711~~ FU-236 -2/C24 (-)	125V VI-BATT BDII PNL 4	FLOOR CLG GLYCOL INLET ISLN VLV
FU-236 -2/C26 (+)	~~52-236 312711~~ FU-236 -2/C26 (-)	125V VI-BATT BDII PNL 4	FLOOR CLG GLYCOL OLET ISLN VLV
FU-236 -2/C34 (+)	~~52-236 312711~~ FU-236 -2/C34 (-)	125V VI-BATT BDII PNL 4	CNTMT BLDG LWR COMPT AIR MON ISLN VLV

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -2/C35 (+)	~~52-236 312711~~ FU-236 -2/C35 (-)	125V VI-BATT BDII PNL 4	CNTMT BLDG UPR COMPT AIR MON ISLN VLV
Insert AA			
FU-236 -2/C41 (+)	~~52-236 312711~~ FU-236 -2/C41 (-)	125V VI-BATT BDII PNL 4	STM GEN NO. 1 BLDN ISLN VLV
FU-236 -2/C42 (+)	~~52-236 312711~~ FU-236 -2/C42 (-)	125V VI-BATT BDII PNL 4	STM GEN NO. 2 BLDN ISLN VLV
FU-236 -2/C43 (+)	~~52-236 312711~~ FU-236 -2/C43 (-)	125V VI-BATT BDII PNL 4	STM GEN NO. 3 BLDN ISLN VLV
FU-236 -2/C44 (+)	~~52-236 312711~~ FU-236 -2/C44 (-)	125V VI-BATT BDII PNL 4	STM GEN NO. 4 BLDN ISLN VLV
FU-236 -2/C47 (+)	~~52-236 312711~~ FU-236 -2/C47 (-)	125V VI-BATT BDII PNL 4	STM GEN BLDN ISLN VLV LP 4
FU-236 -2/D5 (+)	~~52-236 217711~~ FU-236 -2/D5 (-)	125V VI-BATT BDII PNL 4	EXCESS LETDN ISLN VLV
FU-236 -2/D6 (+)	~~52-236 217711~~ FU-236 -2/D6 (-)	125V VI-BATT BDII PNL 4	PRESS RLF TK PR WTR SPLY VLV
FU-236 -2/D7 (+)	~~52-236 217711~~ FU-236 -2/D7 (-)	125V VI-BATT BDII PNL 4	SIS ACC FILL LINE ISLN VLV
FU-236 -2/D8 (+)	~~52-236 217711~~ FU-236 -2/D8 (-)	125V VI-BATT BDII PNL 4	SIS ACC TK 4 N_2 MAKEUP VLV
FU-236 -2/D9 (+)	~~52-236 217711~~ FU-236 -2/D9 (-)	125V VI-BATT BDII PNL 4	SIS ACC TK 4 FLOW ISLN VLV
FU-236 -2/D10 (+)	~~52-236 217711~~ FU-236 -2/D10 (-)	125V VI-BATT BDII PNL 4	SIS ACC TK 4 FILL VLV
Insert BB			
FU-236 -2/D12 (+)	~~52-236 217711~~ FU-236 -2/D12 (-)	125V VI-BATT BDII PNL 4	SIS ACC TK 2 N_2 MAKEUP VLV

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -2/D14 (+)	~~52-236 217/II~~ FU-236 -2/D14(-)	125V VI-BATT BDII PNL 4	SIS ACC TK 2 FLOW ISLN VLV
FU-236 -2/D18 (+)	~~52-236 217/II~~ FU-236 -2/D18(-)	125V VI-BATT BDII PNL 4	RCP 2 SEAL RTN FLOW CONT ~~VLV~~
Insert CC			
FU-236 -2/D21 (+)	~~52-236 217/II~~ FU-236 -2/D21(-)	125V VI-BATT BDII PNL 4	EXCESS LET DN ISLN VLV
FU-236 -2/D22 (+)	~~52-236 217/II~~ FU-236 -2/D22(-)	125V VI-BATT BDII PNL 4	PRESS LIQ ISLN VLV
FU-236 -2/D24 (+)	~~52-236 217/II~~ FU-236 -2/D24(-)	125V VI-BATT BDII PNL 4	RCS HOT LEG LP 3 ISLN VLV
Insert DD			
FU-236 -2/D27 (+)	~~52-236 217/II~~ FU-236 -2/D27(-)	125V VI-BATT BDII PNL 4	SIS ACC TK 4 OUT CK VLV ISLN
Insert EE			
FU-236 -2/D29 (+)	~~52-236 217/II~~ FU-236 -2/D29(-)	125V VI-BATT BDII PNL 4	SIS ACC TK 2 CK VLV ISLN
FU-236 -2/D31 (+)	~~52-236 217/II~~ FU-236 -2/D31(+)	125V VI-BATT BDII PNL 4	SIS CK VLV LEAK TEST ISLN VLV
FU-236 -2/D33 (+)	~~52-236 217/II~~ FU-236 -2/D33(-)	125V VI-BATT BDII PNL 4	SIS TEST LINE CK VLV TEST VLV
FU-236 -2/D34 (+)	~~52-236 217/II~~ FU-236 -2/D34 (-)	125V VI-BATT BDII PNL 4	RCP 2 STD PIPE MAKEUP WTR VLV
FU-236 -2/D35 (+)	~~52-236 217/II~~ FU-236 -2/D35(-)	125V VI-BATT BDII PNL 4	RCP 4 STD PIPE MAKEUP WTR VLV
Insert FF			
FU-236 -2/D44 (+)	~~52-236 217/II~~ FU-236 -2/D44(-)	125V VI-BATT BDII PNL 4	~~PRESS GAS~~ ACCUM TK 1 ~~ISLN VLV~~ ISLN VLV
FU-236 -2/D43 (+)	~~52-236 217/II~~ FU-236 -2/D43(-)	125V VI-BATT BDII PNL 4	RCS HOT LEG LP 1 ISLN VLV

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -2/D45(+)	FU-236 -2/D45(-) ~~52-236 217/11~~	125V VI-BATT BDII PNL 4	ACC TK NO 3 ISLN VLV
Insert GG			
FU-236 -2/E13(+)	FU-236 -2/E13(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	SIS ACC TK 2 FILL VLV
Insert HH			
FU-236 -2/E26(+)	FU-236 -2/E26(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	SIS ACC TK 4 ~~&~~ OTLT CK VLV ISLN VLV
FU-236 -2/E30(+)	FU-236 -2/E30(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	SIS ACC TK 2 ~~&~~ OTLT CK VLV ISLN VLV
FU-236 -2/E32(+)	FU-236 -2/E32(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	RCS LP 2 HOT LEG FEED TEST LINE VLV
Insert II			
FU-236 -2/E37(+)	FU-236 -2/E37(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	NO. 3 SMPL VLV
FU-236 -2/E38(+)	FU-236 -2/E38(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	STM GEN ~~STM GEN 4 DRUM SMPL VLV~~ No.4 SMPL VLV
Insert JJ			
FU-236 -2/E43(+)	FU-236 -2/E43(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	SIS CK VLV LEAK TEST ISLN VLV
FU-236 -2/E44(+)	FU-236 -2/E44(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	REAC VESSEL SEAL LEAK OFF VLV
FU-236 -2/E45(+)	FU-236 -2/E45(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	STM GEN ~~STM GEN~~ NO. 1 BLDN SMPL VLV
FU-236 -2/E46 (+)	FU-236 -2/E46(-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	~~STM GEN~~ NO. 2 BLDN SMPL VLV
FU-236 -2/E47 (+)	FU-236 -2/E47 (-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	~~STM GEN~~ NO. 3 BLDN SMPL VLV
FU-236 -2/E48 (+)	FU-236 -2/E48 (-) ~~52-236 218/11~~	125V VI-BATT BDII PNL 4	~~STM GEN~~ 4 BLDN SMPL VLV

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICE~~S~~	SYSTEM POWERED
5. 125V DC (VI-PWR) (Continued)			
FU-236 -2/E49 (+)	~~52-236 -218/II~~ FU-236 -2/E49 (-)	125V VI-BATT BDII PNL 4	STM GEN NO. 1 SMPL VLV
FU-236 -2/E50 (+)	~~52-236 -218/II~~ FU-236 -2/E50 (-)	125V VI-BATT BDII PNL 4	STM GEN NO. 2 SMPL VLV
~~FU-99 -R55/M18~~	~~52-236 -218/II~~	~~125V VI-BATT BDII PNL 4~~	~~SIS ACC TK 4 FLOW ISLN VLV FCV-63-67~~
~~FU-99 -R55/M18~~	~~52-236 -218/II~~	~~125V VI-BATT BDII PNL 4~~	~~SIS ACC TK 2 FLOW ISLN VLV FCV-63-98~~
~~FU-77 -L2/LF2~~	~~52-236 -214/III~~	~~(PRI) LOCA PNL L-2 125V VI-BATT (BKUP) 125V VI-BATT BD III PNL 2~~	~~RC DR TK TO SUMP TK VLV~~
6. 120VAC BDS			
52-228 -1/2B2	FU-228 -1/~~223~~3	AUX BLDG LTG BD 1	LTG CAB LC-180
52-228 -1/2C2	FU-228 -1/~~225~~4	AUX BLDG LTG BD 1	LTG CAB LC-181
Insert KK ~~52-94 -M18/2~~ ~~FU-F44 A5F4~~ A5F4	52-237 -~~8~~2/M~~D~~18	UNIT CONTROL BD M18 ~~(225) BD M12 (2EXP)~~	TRVLG INCORE PROBE SYS ~~[illegible]~~
~~52-94 -M18/2~~ ~~FU-94 A5F6~~ A5F6	52-237 -4/M7	UNIT CONTROL BD M18 ~~(225) BD M12 (2EXP)~~	TRVLG INCORE PROBE SYS ~~[illegible]~~
~~52-94 -M18/2~~	~~52-237 -4/M7~~	~~UNIT CONTROL BD M18 (228) BD M15 (2EXP)~~	~~TRVLG INCORE PROBE SYS & PATH CONT (G-2)~~

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY DEVICES	SYSTEM POWERED
6. 120 VAC BDS (Continued)			
FU-94 ~~[illegible]~~ -M18A/1 ~~[illegible]~~	FU ~~52~~ -94 ~~[illegible]~~ -M18A/2	UNIT CONTROL BD M18	TRVLG INCORE DEHUMIDIFIER A
Insert LL			
FU-94 ~~[illegible]~~ -M18A/5 ~~[illegible]~~	FU ~~52~~ -94 ~~[illegible]~~ -M18A/6	UNIT CONTROL BD M18	TRVLG INCORE DEHUMIDIFIER B
Insert MM			
FU-94 ~~M18/~~ -M18C/1 ~~[illegible]~~	FU ~~52~~ -94 ~~[illegible]~~ -M18C/2	UNIT CONTROL BD M18	TRVLG INCORE DEHUMIDIFIER C
Insert NN			
FU-94 ~~M18/~~ -M18C/5 ~~[illegible]~~	FU ~~52~~ -94 ~~[illegible]~~ -M18C/6	UNIT CONTROL BD M18	TRVLG INCORE DEHUMIDIFIER ~~D~~ D
Insert OO			
FU-94 ~~M18/~~ -M18D/1 ~~[illegible]~~	FU ~~52~~ -94 ~~[illegible]~~ -M18D/2	UNIT CONTROL BD M18	TRVLG INCORE DEHUMIDIFIER ~~D~~ E
Insert PP			
FU-94 ~~M18/~~ -M18D/5 ~~[illegible]~~	FU ~~52~~ -94 ~~[illegible]~~ M18D/6	UNIT CONTROL BD M18	TRVLG INCORE DEHUMIDIFIER ~~D~~ F
Insert QQ			
FU-278 -M4/~~[illegible]~~ 9, 10	~~52-235~~ ~~FU-278~~ ~~[illegible]~~ ~~M4/10~~ 52-235 -43/II	UNIT CONTROL BD M4 ~~(PRI)~~	RCS LOOP 1 PRES SPRAY CONT VLV POSN INDR
FU-278 -M4/~~[illegible]~~ 11, 12	~~52-235~~ ~~FU-278~~ ~~[illegible]~~ ~~M4/12~~ 52-235 -42/I	UNIT CONTROL BD M4 ~~(PRI)~~	RCS LOOP 2 PRES SPRAY CONT VLV POSN INDR
FU-40 -2272/F3, F4	52 -237 -24/B ~~52-237~~ ~~[illegible]~~ ~~FU-40~~ ~~2272/F4~~	~~UNIT CONTROL BD M7~~ JB 2272	REAC BLDG FLOOD ALARM ~~[illegible]~~
FU-275 -R76/L9, L10	52 -235 ~~52-235~~ ~~[illegible]~~ ~~FU-275~~ ~~R76/[illegible]~~ -12/1-III	AUX RELAY RK 1-R-76	UPR COMPT VT UNIT A FLOW ALA
FU-275 -R76/N21, N22	52 -235 -6/1-III	Aux Relay 1-R-76	TS-30-95, 97 -99, -100

FINAL DRAFT

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	LOCATION OF PRIMARY ~~DEVICES~~ DEVICE	SYSTEM POWERED
6. 120VAC BDS (Continued)			
FU-275 -R76/L15, L16	52-235 ~~52-295~~ ~~FU-275~~ -5/1-IV ~~-5/1-IV~~ ~~-R76/L16~~	AUX RELAY RK 1-R-76	UPR COMPT VT UNIT B FLOW ALA
FU-275 -R76/L7, L8	52-235 ~~52-295~~ ~~FU-275~~ -12/1-III ~~-12/1-III~~ ~~-R76/L8~~	AUX RELAY RK 1-R-76	UPR COMPT VT UNIT C FLOW ALA
FU-275 -R76/L17, L18	52-235 ~~52-295~~ ~~FU-275~~ -5/1-IV ~~-5/1-IV~~ ~~-R76/L18~~	AUX RELAY RK 1-R-76	UPR COMPT VT UNIT D FLOW ALA
FU-99 ~~-R59~~ R58/M19, M20	52-235 ~~52-235~~ ~~FU-99~~ -6/1-IV ~~-6/1-IV~~ ~~-R58/M20~~	NSSS AUX RLY PNL 1-R-58	ICE CONDENSER RELAY
FU-275 -R75/K1, K2	52-235 ~~52-295~~ ~~FU-275~~ -4/1-III ~~-4/1-III~~ ~~-R75/K2~~	AUX RLY PNL 1-R-75	CRD MECH CLR UNIT 1B-B ANN SEP RLY
FU-275 -R75/K21, K22	~~52-295~~ ~~FU-275~~ ~~-8/1-IV~~ ~~-R75/K22~~ 52-235-8/1-IV	AUX RLY PNL 1-R-75	CRD MECH CLR UNIT ~~1A-A~~ 1C-A ANN SEP RLY
~~52-237~~ ~~-1/1~~ FU-242 -BD/F2(+)	~~FU-242~~ ~~-BD/F2~~ FU-242 BD/F2(-)	RAD MON & SMPLG RN DISTR PNL 1	AIR PART MON SYS
Insert (RR)			
7. 250VDC BDS			
FU-202 -A2/1(+)	FU-202 -A2/1(-)	RCP BD 1A/2	RCP MTR 1 PROT
FU-202 -B2/1(+)	FU-202 -B2/1(-)	RCP BD 1B/2	RCP MTR2 PROT
FU-202 -C2/1(+)	FU-202 -C2/1(-)	RCP BD 1C/2	RCP MTR3 PROT
FU-202 -D3/1(+)	FU-202 -D3/1(-)	RCP B 1D/3	RCP MTR4 PROT

INSERT SS

(A)

52-213 -10D/A2	FU-213 A210/32	REAC MOV BD1A2-A	UPR CNTMT VT CLR 1C DISCH ISLN VLV
52-213 -8D/B1	FU-213 -B18/32	REAC MOV BD 1B1-B	SIS ACCUM TK 2 FL ISLN VLV
52-213 -10D/B1	FU-213 -B110/32		RHR SYS ISLN VLV
52-213 -5C/B1	FU-213 -B15/22		RHR SYS ISLN BYP VLV
52-213 -5E/B1	FU-213 -B15/42		RCS PR RELIEF FL CONT VLV
52-213 -6D/B1	FU-213 -B16/32		SEAL FL RET ISLN VLV
52-213 -16A/B1	FU-213 -B116/2		INCORE INST RM CLR FAN 1B
52-213 -16E/B1	FU-213 -B116/52		POWER OUTLETS
52-213 -7D/B1	FU-213 -B17/32		SIS ACCUM TK 4 FL ISLN VLV
52-213 -17E/B1	FU-213 -B117/52		POWER OUTLETS

(B)

52-216 -4F/A	FU-216 -A4/52	FUEL & WASTE HNDLG BD A	FUEL PIT UPENDING WINCH
52-208 -2C/A	FU-208 -A2/22	AUX BLDG COM MCC A	REAC BLDG AUX FLR & EQ SMP PMP 1A
52-208 -5A/A	FU-208 -A5/2		REAC BLDG AUX PMP 1B

(C)

FU
–A17/11A(+)

FU–212
–A17/12N(+)

FU–212
–A17/12A(+)

FU–212
–A17/11A(–)

FU–212
–A17/12 N(–)

FU–212
–A17/12A(–)

480 V SHUT
DN BD 1A1–A

CRD MECH CLR
FAN 1A–A/1

(D)

FU-212 -A17/21A(+)	FU-212 -A17/21A(-)	480 V SHUT DN BD 1A1-A	REAC LWR COMPT CLR FAN 1A-A
FU-212 -A17/22N(+)	FU-212 -A17/22N(-)		
FU-212 -A17/22A(+)	FU-212 -A17/22A(-)		

(E)

FU-212 -A110/22(+)	FU-212 -A110/22(-)	480 V SHUT DN BD 1A1-A	CNTMNT AIR RTN FAN 1A-A

(F)

FU-212
-A27/1A(+)

FU-212
-A27/2N(+)

FU-212
-A27/2A(+)

FU-212
-A27/1A(-)

FU-212
-A27/2N(-)

FU-212
-A27/2N(-)

480 V SHUT
DN BD 1A2-A

CRD MECH CLR
FAN 1C-A/1

(G)

FU–212 –A27/31A(+)	FU–212 –A27/31A(–)	480 V SHUT DN BD 1A2–A	REAC LWR COMPT CLR FAN 1C–A
FU–212 –A27/32N(+)	FU–212 –A27/32N(–)		
FU–212 –A27/32A(+)	FU–212 –A27/32A(–)		

(H)

FU-212 -B17/31A(+)	FU-212 -B17/31A(-)	480V SHUT DN BD 1B1-B	REAC LWR COMPT CLR FAN 1B-B
FU-212 -B17/32N(+)	FU-212 -B17/32N(-)		
FU-212 -B17/32A(+)	FU-212 -B17/32A(-)		

(I)

FU-212 -B17/21A(+)	FU-212 -B17/21A(-)	480 V SHUT DN BD 1B1-B	CRD MECH CLR FAN 1B-B/1
FU-212 -B17/22N(+)	FU-212 -B17/22N(-)		
FU-212 -B17/22A(+)	FU-212 -B17/22A(-)		

(J)

FU-212 -B27/11A(+)	FU-212 -B27/11A(-)	480 V SHUT DN BD 1B2-B	CRD MECH CLR FAN 1D-B/1
FU-212 -B27/12N(+)	FU-212 -B27/12N(-)		
FU-212 -B27/12A(-)	FU-212 -B27/12A(-)		

(K)

FU-212 -B27/31A(+)	FU-212 -B27/31A(-)	480V SHUT DN BD 1B2-B	REAC LWR COMPT CLR FAN 1 D-B
FU-212 -B27/32N(+)	FU-212 -B27/32N(-)	↓	↓
FU-212 -B27/32A(+)	FU-212 -B27/32A(-)		

(L)

FU-212 -B29/22(+)	FU-212 -B29/22(-)	480V SHUT DN BD 1B2-B	CNTMT AIR RTN FAN 1B-B

(M)

FU-236 -1/A5(+)	FU-236 -1/A5(-)	125V VI-BATT BD I PNL 4	CHRG FLOW TO RCS SPRAY

(N)

FU-236 -1/A18(+)	FU-236 -1/A18(-)	125V VI-BATT BD I PNL 4	REAC VES HD VT ISLN VLV & REAC VES HD VT THROTTLE VLV

FU-236 -1/B1(+)	FU-236 -1/B1(-)	125V V1-BATT BD 1 PNL 4	CHG FL TO RCS COOL LP 4
FU-236 -1/B3(+)	FU-236 -1/B3(-)	125V V1-BATT BD 1 PNL 4	RCP LP 3 LT DN FLOW

(P)

FU-236 -1/B6(+)	FU-236 -1/B6(-)	125V VI-BATT BD I PNL 4	REGEN HT EXCH LT DN ISLN VLV A
FU-236 -1/B7(+)	FU-236 -1/B7(-)		REGEN HT EXCH LT DN ISLN VLV B
FU-236 -1/B8(+)	FU-236 -1/B8(-)		REGEN HT EXCH LT DN ISLN VLV C
FU-236 -1/B9(+)	FU-236 -1/B9(-)		RCP LP 3 LT DN FLOW
FU-236 -1/B11(+)	FU-236 -1/B11(-)		RCS PRESS PWR RLF VLV

FU-236 -1/B19(+)	FU-236 -1/B19(-)	125V VI-BATT BDI -PNL 4	SIS CK VLV ISLN HDR FLOW ISLN VLV
FU-236 -1/B21(+)	FU-236 -1/B21(-)		LWR CNTMT VT CLR A SPLY VLV
FU-236 -1/B22(+)	FU-236 -1/B22(-)		CRD VT CLR A SPLY VLV
FU-236 -1/B23(+)	FU-236 -1/B23(-)		LWR CNTMT VT CLR C SPLY VLV
FU-236 -1/B24(+)	FU-236 -1/B24(-)		CRD VT CLR C SPLY VLV

(R)

~~FU-236 -1/B31(+)~~ ~~FU-236 -1/B31(-)~~ ~~125V VI-BATT BD I PNL4~~ ~~SIS CK VLV ISLN HDR FL ISLN VLV~~

INSERT R is deleted

(S)

FU-236 -1/B40(+)	FU-236 -1/B40(-)	125V VI-BATT BD 1 PNL 4	RCP MTR CLR A SUPP VLV
FU-236 -1/B41(+)	FU-236 -1/B41(-)		RCP MTR CLR C SPLY VLV
FU-236 -1/B42(+)	FU-236 -1/B42(-)		CRD CLG UNIT 1A-A SUCT DMPR
FU-236 -1/B43(+)	FU-236 -1/B43(-)		CRD CLG UNIT 1A-A RM DIVR DMPR
FU-236 -1/B44(+)	FU-236 -1/B44(-)		CRD CLG UNIT 1C-A SUCT DMPR
FU-236 -1/B45(+)	FU-236 -1/B45(-)		CRD CLG UNIT 1C-A RM DIVR DMPR

(T)

FU-236 -1/D7(+)	FU-236 -1/D7(-)	125V VI-BATT BD I PNL 4	SIS ACCUM TK 1 OTLT DRAIN VLV
FU-236 -1/D12(+)	FU-236 -1/D12(-)		SIS ACCUM TK 3 OTLT CK VLV ISLN VLV
FU-236 -1/D14(+)	FU-236 -1/D14(-)		NO.1 SEAL BYP FL CNTL VLV

(U)

FU-236 -1/D20(+)	FU-236 -1/D20(-)	125V VI-BATT BD 1 PNL 4	RCS PRESS RLF TK VT
FU-236 -1/D21(+)	FU-236 -1/D21(-)		RCS PRESS RLF TK DR

(V)

FU-236 -1/D23(+)	FU-236 -1/D23(-)	125V VI-BATT BD 1 PNL 4	SIS ACCUM TK 3 FILL VLV
FU-236 -1/D24(+)	FU-236 -1/D24(-)		SIS ACCUM TK 1 FILL VLV
FU-236 -1/D27(+)	FU-236 -1/D27(-)		RHR SPLY TEST LINE VLV
FU-236 -1/D28(+)	FU-236 -1/D28(-)		SIS PMP OTLT TO SIS TEST LINE
FU-236 -1/D29(+)	FU-236 -1/D29(-)		SIS CLR 3 CK VLV ISLN
FU-236 -1/D30(+)	FU-236 -1/D30(-)		SIS ACCUM TK1 N_2 MKUP VLV
FU-236 -1/D31(+)	FU-236 -1/D31(-)		RCS LP 1 HT LEG FD TEST LINE VLV
FU-236 -1/D32(+)	FU-236 -1/D32(-)		SIS FL TO COLD LEG CK VLV TEST

(W)

FU-236 -1/D37(+)	FU-236 -1/D37(-)	125V VI-BATT BD 1 PNL 4	SIS PMP OTLT TEST LINE VLV
FU-236 -1/D38(+)	FU-236 -1/D38(-)		RHR RTN FROM SIS SMPL LINE VLV
FU-236 -1/D40(+)	FU-236 -1/D40(-)		RHR SPLY TEST LINE VLV

FU-236 -1/D42(+)	FU-236 -1/D42(-)	125V VI-BATT BD 1 PNL 4	RHR SPLY TEST LINE VLV
FU-236 -1/D45(+)	FU-236 -1/D45(-)		RHR SPLY TEST LINE VLV
FU-236 -1/E2(+)	FU-236 -1/E2(-)		RCP 3 SEAL RTN FL CONT
FU-236 -1/E7(+)	FU-236 -1/E7(-)		SIS ACCUM TK 1 OTLT FL ISLN VLV
FU-236 -1/E14(+)	FU-236 -1/E14(-)		NO.1 SEAL BYP FL CNTL VLV
FU-236 -1/E17(+)	FU-236 -1/E17(-)		RCP 1 SEAL RTN FL CONT
FU-236 -1/E20(+)	FU-236 -1/E20(-)		RCS PRESS RLF TK VT VLV
FU-236 VLV -1/E21(+)	FU-236 -1/E21(-)		RCS PRESS RLF TK VT
FU-236 -1/E23(+)	FU-236 -1/E23(-)		SIS ACCUM TK 3 FILL VLV
FU-236 -1/E24(+)	FU-236 -1/E24(-)		SIS ACCUM TK 1 FILL VLV
FU-236 -1/E30(+)	FU-236 -1/E30(-)		SIS ACCUM TK 1 N_2 MKUP VLV
FU-236 -1/E32(+)	FU-236 -1/E32(-)		SIS ACCUM TK 1 OTLT CK VLV ISLN VLV

(Y)

FU-236 -2/A23(+)	FU-236 -2/A23(-)	125V VI-BATT BD 2 PNL 4	ACCUM TKS ISLN VLV
FU-236 -2/A24(+)	FU-236 -2/A24(-)		REAC CLT DRN TK TO GAS ANAL ISLN VLV
FU-236 -2/A41(+)	FU-236 -2/A41(-)		RCP MTR CLR B SUP VLV
FU-236 -2/A42(+)	FU-236 -2/A42(-)		RCP MTR CLR D SUP VLV
FU-236 -2/A43(+)	FU-236 -2/A43(-)		REAC BLDG SMP PMP DISCH ISLN VLV
FU-236 -2/A44(+)	FU-236 -2/A44(-)		REAC CLT DR TK PMPS DISCH ISLN VLV
FU-236 -2/A45(+)	FU-236 -2/A45(-)		REAC CLT DR TK TO VT HDR ISLN VLV
FU-236 -2/B3(+)	FU-236 -2/B3(-)		CHGR FL RCS CL LP 1
FU-236 -2/B4(+)	FU-236 -2/B4(-)		RES FLOW CONT VLV
FU-236 -2/B5(+)	FU-236 -2/B5(-)		RCS PRESS PWR RLF VLV
FU-236 -2/B8(+)	FU-236 -2/B8(-)		EXCESS LTDN DIVR FL CONT VLV
FU-236 -2/B9(+)	FU-236 -2/B9(-)		LWR CNTMT VT CLR B SPLY VLV
FU-236 -2/B10(+)	FU-236 -2/B10(-)		CRD VT CLR B SPLY VLV
FU-236 -2/B11(+)	FU-236 -2/B11(-)		LWR CNTMT VT CLR D SPLY VLV

(Y)

FU-236 -2/B12(+)	FU-236 -2/B12(-)	125V VI-BATT BD 2 PNL 4	CRD VT CLR D SPLY VLV
FU-236 -2/B13(+)	FU-236 -2/B13(-)		CRD CLG UNIT 1D-B RM DIVR DMPR
FU-236 -2/B14(+)	FU-236 -2/B14(-)		CRD CLG UNIT 1D-B RM DIVR DMPR
FU-236 -2/B15(+)	FU-236 -2/B15(-)		CRD CLG UNIT 1B-B SUCT DMPR
FU-236 -2/B16(+)	FU-236 -2/B16(-)		CRD CLG UNIT 1B-B DIVR DMPR
FU-236 -2/B17(+)	FU-236 -2/B17(-)		REAC VSL HD VT ISLN VLV & THROT VLV
FU-236 -2/B21(+)	FU-236 -2/B21(-)		REAC BLDG SMP DISCH ISLN VLV
FU-236 -2/B22(+)	FU-236 -2/B22(-)		REAC CLT DR TK PMPS DISCH ISLN VLV
FU-236 -2/B23(+)	FU-236 -2/B23(-)		REAC CLT DR TK TO VT HDR ISLN VLV
FU-236 -2/B24(+)	FU-236 -2/B24(-)		REAC CLT DRN TK TO GAS ANAL ISLN VLV
FU-236 -2/B27(+)	FU-236 -2/B27(-)		CNTMT BLDG LWR COMPT AIR MON ISLN VLV
FU-236 -2/B28(+)	FU-236 -2/B28(-)		CNTMT BLDG LWR COMPT AIR MON ISLN VLV

(Z)

FU-236 -2/C18(+)	FU-236 -2/C18(-)	125V VI-BATT BD 2 PNL 4	UPR COMPT PURGE ISLN VLV & EXH ISLN VLV

(AA)

FU-236 -2/C40(+)	FU-236 -2/C40(-)	125V VI-BATT BD 2 PNL 4	LWR COMPT PURGE ISLN VLVS
FU-236 -2/C37(+)	FU-236 -2/C37(-)	125V VI-BATT BD 2 PNL 4	INSTR RM PURGE ISLN VLVS

(BB)

FU-236 -2/D11(+)	52-236 -2/D11(-)	125V VI-BATT BD 2 PNL 4	SIS ACCUM TK3 N_2 MKUP VLV

(CC)

FU-236 -2/D19(+)	FU-236 -2/D19(-)	125V VI-BATT BD 2 PNL 4	RCP 4 SEAL RTN FL CONT

(DD)

FU-236 -2/D26(+)	FU-236 -2/D26(-)	125V VI-BATT BD 2 PNL4	ACCUM TK 2 ISLN VLV

(EE)

FU-236 -2/D28(+)	FU-236 -2/D28(-)	125V VI-BATT BD 2 PNL 4	ACCUM TK 4 ISLN VLV

(FF)

FU-236 -2/D36(+)	FU-236 -2/D36(-)	125V VI-BATT BD 2 PNL 4	SIS ACCUM TK 3 FL ISLN VLV
FU-236 -2/D40(+)	FU-236 -2/D40(-)		SIS ACCUM TK 2 FILL VLV
FU-236 -2/D41(+)	FU-236 -2/D41(-)		PRESS GAS ISLN VLV

(GG)

FU-236 -2/E5(+)	FU-236 -2/E5(-)	125V VI-BATT BD 2 PNL 4	EXCESS LTDN ISLN VLV
FU-236 -2/E6(+)	FU-236 -2/E6(-)		PRESS RLF TK PR WTR SPLY VLV
FU-236 -2/E7(+)	FU-236 -2/E7(-)		SIS ACCUM FILL LINE ISLN VLV
FU-236 -2/E8(+)	FU-236 -2/E8(-)		SIS ACCUM TK 4 N_2 MKUP VLV
FU-236 -2/E9(+)	FU-236 -2/E9(-)		SIS ACCUM TK 4 FL ISLN VLV
FU-236 -2/E10(+)	FU-236 -2/E10(-)		SIS ACCUM TK 4 FILL VLV
FU-236 -2/E11(+)	FU-236 -2/E11(-)		SIS ACCUM TK 3 N_2 MKUP VLV

(HH)

FU-236 -2/E14(+)	FU-236 -2/E14(-)	125V VI-BATT BD 2 PNL 4	SIS ACCUM TK 2 N_2 MKUP VLV
FU-236 -2/E15(+)	FU-236 -2/E15(-)		SIS ACCUM TK 2 FL ISLN VLV
FU-236 -2/E18(+)	FU-236 -2/E18(-)		RCP 2 SEAL RTN FL CONT
FU-236 -2/E19(+)	FU-236 -2/E19(-)		RCP 4 SEAL RTN FL CONT
FU-236 -2/E20(+)	FU-236 -2/E20(-)		RCS LP 4 HOT LEG FD TEST LINE VLV
FU-236 -2/E21(+)	FU-236 -2/E21(-)		EXCESS LETDOWN ISLN VLV

(II)

FU-236 -2/E33(+)	FU-236 -2/E33(-)	125V VI-BATT BD 2 PNL 4	SIS ACCUM TK 1 OTLT CK VLV ISLN VLV

(JJ)

FU-236 -2/E39(+)	FU-236 -2/E39(-)	125V VI-BATT BD 2 PNL 4	SIS ACCUM TK 3 FL ISLN VLV

(KK)

52-268 -1/1A	FU-268 -1/1A	HY MIT DIST PNL 1A	IGNITORS 1 and 32
52-268 -2/1A	FU-268 -2/1A		IGNITORS 2 and 23
52-268 -3/1A	FU-268 -3/1A		IGNITORS 5 and 6
52-268 -4/1A	FU-268 -4/1A		IGNITORS 7 and 8
52-268 -5/1A	FU-268 -5/1A		IGNITORS 13 and 14
52-268 -6/1A	FU-268 -6/1A		IGNITORS 15 and 16
52-268 -7/1A	FU-268 -7/1A		IGNITORS 21 and 22
52-268 -8/1A	FU-268 -8/1A		IGNITORS 25 and 53
52-268 -9/1A	FU-268 -9/1A		IGNITORS 24 and 29
52-268 -10/1A	FU-268 -10/1A		IGNITORS 30 and 31
52-268 -11/1A	FU-268 -11/1A		IGNITORS 35 and 36
52-268 -12/1A	FU-268 -12/1A		IGNITORS 26 and 33
52-268 -13/1A	FU-268 -13/1A		IGNITORS 34 and 42
52-268 -14/1A	FU-268 -14/1A		IGNITORS 49 and 50

(KK)

52-268 -15/1A	FU-268 -15/1A	HY MIT DIST PNL 1A	IGNITORS 54 and 55
52-268 -16/1A	FU-268 -16/1A		IGNITORS 27 and 28
52-268 -17/1A	FU-268 -17/1A		IGNITORS 65 and 66
52-268 -1/2B	FU-268 -1/2B	HY MIT DIST PNL 2B	IGNITORS 3 and 48
52-268 -2/2B	FU-268 -2/2B		IGNITORS 4 and 37
52-268 -3/2B	FU-268 -3/2B		IGNITORS 9 and 10
52-268 -4/2B	FU-268 -4/2B		IGNITORS 11 and 12
52-268 -5/2B	FU-268 -5/2B		IGNITORS 17 and 18
52-268 -6/2B	FU-268 -6/2B		IGNITORS 19 and 20
52-268 -7/2B	FU-268 -7/2B		IGNITORS 38 and 39
52-268 -8/2B	FU-268 -8/2B		IGNITORS 43 and 44
52-268 -9/2B	FU-268 -9/2B		IGNITORS 41 and 59
52-268 -10/2B	FU-268 -10/2B		IGNITORS 45 and 46
52-268 -11/2B	FU-268 -11/2B		IGNITORS 40 and 47

(KK)

52-268 -12/2B	FU-268 -12/2B	HY MIT DIST PNL 2B	IGNITORS 51 and 52
52-268 -13/2B	FU-268 -13/2B		IGNITORS 56 and 57
52-268 -14/2B	FU-268 -14/2B		IGNITORS 58 and 60
52-268 -15/2B	FU-268 -15/2B		IGNITORS 61 and 62
52-268 -16/2B	FU-268 -16/2B		IGNITORS 63 and 64
52-268 -17/2B	FU-268 -17/2B		IGNITORS 67 and 68
FU-85 -L117/13(+)	FU-85 -L117/17(-)	PNL 1-L-117	CRDM B6 STA
FU-85 -L117/14(+)	FU-85 -L117/18(-)		CRDM F14 STA
FU-85 -L117/15(+)	FU-85 -L117/19(-)		CRDM P10 STA
FU-85 -L117/16(+)	FU-85 -L117/20(-)		CRDM K2 STA
FU-85 -L117/21(+)	FU-85 -L117/45(-)		CRDM B6 MOV
FU-85 -L117/22(+)	FU-85 -L117/46(-)		CRDM F14 MOV
FU-85 -L117/23(+)	FU-85 -L117/47(-)		CRDM P10 MOV
FU-85 -L117/24(+)	FU-85 -L117/48(-)		CRDM K2 MOV

(KK)

FU-85 -L117/25(+)	FU-85 -L117/29(-)	PNL 1-L-117	CRDM H4 STA
FU-85 -L117/26(+)	FU-85 -L117/30(-)		CRDM D8 STA
FU-85 -L117/27(-)	FU-85 -L117/31(-)		CRDM H12 STA
FU-85 -L117/28(+)	FU-85 -L117/32(-)		CRDM M8 STA
FU-85 -L117/33(+)	FU-85 -L117/45(-)		CRDM H4 MOV
FU-85 -L117/34(+)	FU-85 -L117/46(-)		CRDM D8 MOV
FU-85 -L117/35(+)	FU-85 -L117/47(-)		CRDM H12 MOV
FU-85 -L117/36(+)	FU-85 -L117/48(-)		CRDM M8 MOV
FU-85 -L117/41(+)	FU-85 -L117/37(-)		CRDM C7 STA
FU-85 -L117/42(+)	FU-85 -L117/38(-)		CRDM G13 STA
FU-85 -L117/43(+)	FU-85 -L117/39(-)		CRDM N9 STA
FU-85 -L117/44(+)	FU-85 -L117/40(-)		CRDM J3 STA
FU-85 -L117/49(+)	FU-85 -L117/45(-)		CRDM C7 STA
FU-85 -L117/50(+)	FU-85 -L117/46(-)		CRDM G13 MOV

(KK)

FU-85 -L117/80(+)	FU-85 -L117/84(-)	PNL 1-L-117	CRDM N9 LIFT
FU-85 -L117/81(+)	FU-85 -L117/85(-)		CRDM J3 LIFT
FU-85 -L117/86(+)	FU-85 -L117/87(-)		CRDM H8 LIFT
FU-85 -L118/13(+)	FU-85 -L117/17(-)	PNL 1-L-118	CRDM F8 STA
FU-85 -L118/14(+)	FU-85 -L118/18(-)		CRDM K8 STA
FU-85 -L118/21(+)	FU-85 -L118/45(-)		CRDM F8 MOV
FU-85 -L118/22(+)	FU-85 -L118/46(-)		CRDM K8 MOV
FU-85 -L118/25(+)	FU-85 -L118/29(-)		CRDM F6 STA
FU-85 -L118/26(+)	FU-85 -L118/30(-)		CRDM F10 STA
FU-85 -L118/27(+)	FU-85 -L118/31(-)		CRDM K10 STA
FU-85 -L118/28(+)	FU-85 -L118/32(-)		CRDM K6 STA
FU-85 -L118/33(+)	FU-85 -L118/45(-)		CRDM F6 MOV
FU-85 -L118/34(+)	FU-85 -L118/46(-)		CRDM F10 MOV
FU-85 -L118/35(+)	FU-85 -L118/47(-)		CRDM K10 MOV

(KK)

FU-85 -L118/36(+)	FU-85 -L118/48(-)	PNL 1-L-118	CRDM K6 MOV
FU-85 -L118/41(+)	FU-85 -L118/37(-)		CRDM B4 STA
FU-85 -L118/42(+)	FU-85 -L118/38(-)		CRDM D14 STA
FU-85 -L118/43(+)	FU-85 -L118/39(-)		CRDM P12 STA
FU-85 -L118/44(+)	FU-85 -L118/40(-)		CRDM M2 STA
FU-85 -L118/49(+)	FU-85 -L118/45(-)		CRDM B4 MOV
FU-85 -L118/50(+)	FU-85 -L118/46(-)		CRDM D14 MOV
FU-85 -L118/51(+)	FU-85 -L118/47(-)		CRDM P12 MOV
FU-85 -L118/52(+)	FU-85 -L118/48(-)		CRDM M2 MOV
FU-85 -L118/70(+)	FU-85 -L118/82(-)		CRDM F8 LIFT
FU-85 -L118/71(+)	FU-85 -L118/83(-)		CRDM K3 LIFT
FU-85 -L118/74(+)	FU-85 -L118/82(-)		CRDM F6 LIFT
FU-85 -L118/75(+)	FU-85 -L118/83(-)		CRDM F10 LIFT
FU-85 -L118/76(+)	FU-85 -L118/84(-)		CRDM K10 LIFT

(KK)

FU-85 -L118/77(+)	FU-85 -L118/85(-)	PNL 1-L-118	CRDM K6 LIFT
FU-85 -L118/78(+)	FU-85 -L118/82(-)		CRDM B4 LIFT
FU-85 -L118/79(+)	FU-85 -L118/83(-)		CRDM D14 LIFT
FU-85 -L118/80(+)	FU-85 -L118/84(-)		CRDM P12 LIFT
FU-85 -L118/81(+)	FU-85 -L118/85(-)		CRDM M2 LIFT
FU-85 -L119/13(+)	FU-85 -L119/17(-)	PNL 1-L-119	CRDM E3 STA
FU-85 -L119/14(+)	FU-85 -L119/18(-)		CRDM C11 STA
FU-85 -L119/15(+)	FU-85 -L119/19(-)		CRDM L13 STA
FU-85 -L119/16(+)	FU-85 -L119/20(-)		CRDM N5 STA
FU-85 -L119/21(+)	FU-85 -L119/45(-)		CRDM E3 MOV
FU-85 -L119/22(+)	FU-85 -L119/46(-)		CRDM C11 MOV
FU-85 -L119/23(+)	FU-85 -L119/47(-)		CRDM L13 MOV
FU-85 -L119/24(+)	FU-85 -L119/48(-)		CRDM N5 MOV
FU-85 -L119/25(+)	FU-85 -L119/29(-)		CRDM C5 STA

(KK)

FU-85 -L119/26(+)	FU-85 -L119/30(-)	PNL 1-L-119	CRDM E13 STA
FU-85 -L119/27(+)	FU-85 -L119/31(-)		CRDM N11 STA
FU-85 -L119/28(+)	FU-85 -L119/32(-)		CRDM L3 STA
FU-85 -L119/33(+)	FU-85 -L119/45(-)		CRDM C5 MOV
FU-85 -L119/34(+)	FU-85 -L119/46(-)		CRDM E13 MOV
FU-85 -L119/35(+)	FU-85 -L119/47(-)		CRDM N11 MOV
FU-85 -L119/36(+)	FU-85 -L119/48(-)		CRDM L3 MOV
FU-85 -L119/70(+)	FU-85 -L119/82(-)		CRDM E3 LIFT
FU-85 -L119/71(+)	FU-85 -L119/83(-)		CRDM C11 LIFT
FU-85 -L119/72(+)	FU-85 -L119/84(-)		CRDM L13 LIFT
FU-85 -L119/73(+)	FU-85 -L119/85(-)		CRDM N5 LIFT
FU-85 -L119/74(+)	FU-85 -L119/82(-)		CRDM C5 LIFT
FU-85 -L119/75(+)	FU-85 -L119/83(-)		CRDM E13 LIFT
FU-85 -L119/76(+)	FU-85 -L119/84(-)		CRDM N11 LIFT

(KK)

FU-85 -L119/77(+)	FU-85 -L119/85(-)	PNL 1-L-119	CRDM L3 LIFT
FU-85 -L120/13(+)	FU-85 -L120/17(-)	PNL 1-L-120	CRDM H6 STA
FU-85 -L120/14(+)	FU-85 -L120/18(-)		CRDM H10 STA
FU-85 -L120/21(+)	FU-85 -L120/45(-)		CRDM H6 MOV
FU-85 -L120/22(+)	FU-85 -L120/46(-)		CRDM H10 MOV
FU-85 -L120/25(+)	FU-85 -L120/29(-)		CRDM H2 STA
FU-85 -L120/26(+)	FU-85 -L120/30(-)		CRDM B8 STA
FU-85 -L120/27(+)	FU-85 -L120/31(-)		CRDM H14 STA
FU-85 -L120/28(+)	FU-85 -L120/32(-)		CRDM P8 STA
FU-85 -L120/33(+)	FU-85 -L120/45(-)		CRDM H2 MOV
FU-85 -L120/34(+)	FU-85 -L120/46(-)		CRDM B8 MOV
FU-85 -L120/35(+)	FU-85 -L120/47(-)		CRDM H14 MOV
FU-85 -L120/36(+)	FU-85 -L120/48(-)		CRDM P8 MOV
FU-85 -L120/41(+)	FU-85 -L120/37(-)		CRDM D2 STA

(KK)

FU-85 -L120/42(+)	FU-85 -L120/38(-)	PNL 1-L-120	CRDM B12 STA
FU-85 -L120/43(+)	FU-85 -L120/39(-)		CRDM M14 STA
FU-85 -L120/44(+)	FU-85 -L120/40(-)		CRDM P4 STA
FU-85 -L120/49(+)	FU-85 -L120-/45(-)		CRDM D2 MOV
FU-85 -L120/50(+)	FU-85 -L120/46(-)		CRDM B12 MOV
FU-85 -L120/51(+)	FU-85 -L120/47(-)		CRDM M14 MOV
FU-85 -L120/52(+)	FU-85 -L120/48(-)		CRDM P4 MOV
FU-85 -L120/70(+)	FU-85 -L120/82(-)		CRDM H6 LIFT
FU-85 -L120/71(+)	FU-85 -L120/83(-)		CRDM H10 LIFT
FU-85 -L120/74(+)	FU-85 -L120/82(-)		CRDM H2 LIFT
FU-85 -L120/75(+)	FU-85 -L120/83(-)		CRDM B8 LIFT
FU-85 -L120/76(+)	FU-85 -L120/84(-)		CRDM H14 LIFT
FU-85 -L120/77(+)	FU-85 -L120/85(-)		CRDM P8 LIFT
FU-85 -L120/78(+)	FU-85 -L120/82(-)		CRDM D2 LIFT

(KK)

FU-85 -L120/79(+)	FU-85 -L120/83(-)	PNL 1-L-120	CRDM B12 LIFT
FU-85 -L120/80(+)	FU-85 -L120/84(-)		CRDM M14 LIFT
FU-85 -L120/81(+)	FU-85 -L120/85(-)		CRDM P4 LIFT
FU-85 -L121/13(+)	FU-85 -L121/17(-)	PNL 1-L-121	CRDM F2 STA
FU-85 -L121/14(+)	FU-85 -L121/18(-)		CRDM B10 STA
FU-85 -L121/15(+)	FU-85 -L121/19(-)		CRDM K14 STA
FU-85 -L121/16(+)	FU-85 -L121/20(-)		CRDM P6 STA
FU-85 -L121/21(+)	FU-85 -L121/45(-)		CRDM F2 MOV
FU-85 -L121/22(+)	FU-85 -L121/46(-)		CRDM B10 MOV
FU-85 -L121/23(+)	FU-85 -L121/47(-)		CRDM K14 MOV
FU-85 -L121/24(+)	FU-85 -L121/48(-)		CRDM P6 MOV
FU-85 -L121/25(+)	FU-85 -L121/29(-)		CRDM D4 STA
FU-85 -L121/26(+)	FU-85 -L121/30(-)		CRDM D12 STA
FU-85 -L121/27(+)	FU-85 -L121/31(-)		CRDM M12 STA

(KK)

FU-85 -L121/28(+)	FU-85 -L121/32(-)	PNL 1-L-121	CRDM M4 STA
FU-85 -L121/33(+)	FU-85 -L121/45(-)		CRDM D4 MOV
FU-85 -L121/34(+)	FU-85 -L121/45(-)		CRDM D12 MOV
FU-85 -L121/35(+)	FU-85 -L121/47(-)		CRDM M12 MOV
FU-85 -L121/36(+)	FU-85 -L121/48(-)		CRDM M4 MOV
FU-85 -L121/41(+)	FU-85 -L121/37(-)		CRDM G3 STA
FU-85 -L121/42(+)	FU-85 -L121/38(-)		CRDM C9 STA
FU-85 -L121/43(+)	FU-85 -L121/39(-)		CRDM J13 STA
FU-85 -L121/44(+)	FU-85 -L121/40(-)		CRDM N7 STA
FU-85 -L121/49(+)	FU-85 -L121/45(-)		CRDM G3 MOV
FU-85 -L121/50(+)	FU-85 -L121/46(-)		CRDM C9 MOV
FU-85 -L121/51(+)	FU-85 -L121/47(-)		CRDM J13 MOV
FU-85 -L121/52(+)	FU-85 -L121/48(-)		CRDM N7 MOV
FU-85 -L121/70(+)	FU-85 -L121/82(-)		CRDM F2 LIFT

(KK)

FU-85 -L121/71(+)	FU-85 -L121/83(-)	PNL 1-L-121	CRDM B10 LIFT
FU-85 -L121/72(+)	FU-85 -L121/84(-)		CRDM K14 LIFT
FU-85 -L121/73(+)	FU-85 -L121/85(-)		CRDM P6 LIFT
FU-85 -L121/74(+)	FU-85 -L121/82(-)		CRDM D4 LIFT
FU-85 -L121/75(+)	FU-85 -L121/83(-)		CRDM D12 LIFT
FU-85 -L121/76(+)	FU-85 -L121/84(-)		CRDM M12 LIFT
FU-85 -L121/77(+)	FU-85 -L121/85(-)		CRDM M4 LIFT
FU-85 -L121/78(+)	FU-85 -L121/82(-)		CRDM G3 LIFT
FU-85 -L121/79(+)	FU-85 -L121/83(-)		CRDM C9 LIFT
FU-85 -L121/80(+)	FU-85 -L121/84(-)		CRDM J13 LIFT
FU-85 -L121/81(+)	FU-85 -L121/85(-)		CRDM N7 LIFT

(LL)

FU-94 -M18A/3	FU-94 -M18A/4	UNIT CONT BD M18	TRVLG INCORE DEHUMIDIFIER A

(LL)

FU-94 -M18A/3	FU-94 -M18A/4	UNIT CONT BD M18	TRVLG INCORE DEHUMIDIFIER A

(MM)

FU-94 -M18A/7	FU-94 -M18A/8	UNIT CONT BD M18	TRVLG INCORE DEHUMIDIFIER B

(NN)

FU-94 -M18C/3	FU-M18 -M18C/4	UNIT CONT BD M18	TRVLG INCORE DEHUMIDIFIER C

(00)

FU-94 -M18C/7	FU-94 -M18C/8	UNIT CONT BD M18	TRVLG INCORE DEHUMIDIFIER D

(PP)

FU-94 -M18D/3	FU-94 -M18D/4	UNIT CONT BD M18	TRVLG INCORE DEHUMIDIFIER E

(QQ)

FU-94 -M18D/7	FU-94 -M18D/8	UNIT CONT BD M18	TRVLG INCORE DEHUMIDIFIER F

(RR)

52-242 -11/1	FU-242 -DPL1/F11	RAD PROC AND AREA MON PWR DIST PNL1	UPR COMPT ACC HATCH AREA MON
52-242 -12/1	FU-242 -DPL1/F12	RAD PROC AND AREA MON PWR DIST PNL1	UPR COMPT PERS LOCK AREA MON
52-242 -13/1	FU-242 -DPL1/F13	RAD PROC AND AREA MON PWR DIST PNL1	LWR COMPT INSTR RM AREA MON
FU-99 -R58/M7	FU-99 -R58/M8	AUX RLY PNL 1-R-58	COMTMT EVAC HORN
FU-237 -M7B/F3	FU-237 -M7B/F4	UNIT CONT BD M7	REAC COOL ST AND FW
FU-275 -R75/L11, L12	52-235 -4	AUX RLY PNL 1-R-75	CRD MECH CLR FAN ID-B/1 ANN SEP RLY
FU-275 -R75/K19, K20	52-235 -8	AUX RLY PNL 1-R-75	CRD MECH CLR FAN 1A-A/1 ANN SEP RLY
FU-234 -A1/1	52-234 -2/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 132P
FU-234 -B1/4	52-234 -28/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 214S
FU-234 -B1/1	52-234 -22/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 132S
FU-234 -A1/2	52-234 -24/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 213P
FU-234 -B1/5	52-234 -2/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 134S
FU-234 -B1/16	52-234 -30/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 215S
FU-234 -B1/7	52-234 -32/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 135S
FU-234 -B1/8	52-234 -34/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 216S

(RR)

FU-234 -B1/2	52-234 -24/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 213S
FU-234 -A1/3	52-234 -26/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 133P
FU-234 -B1/3	52-234 -26/B1	SIS HT TR DIST PNL B1	HT TRACE CKT 133S
FU-234 -A1/4	52-234 -28/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 214P
FU-275 -R79/K11, K12	52-237 15/M7B	AUX RLY PNL 1-R-79	FIRE PMP ST CIRCUIT
FU-275 -R58/M19, M20	52-235 -6/IV	AUX RLY PNL 1-R-58	GLY EXP TK LVL SW
FU-237 -A/F23	52-237 -23/1A	120 VAC DIST PNL 1A	I CBK DRAFT DMPR
FU-292 -4314/FS55,FS56	52-235 -12/II	JB 4314	INCORE INST RM CLR FAN 1A
FU-292 -4313/FS53, FS54	52-235 -7/I	JB 4313	INCORE INST RM A/C PKG CLR FAN 1B
FU-275 -R75/L9	FU-275 -R75/L10	AUX PLY PNL 1-R-75	AUX BLDG EGTS FAN B-B
FU-234 -A1/5	52-234 -2/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 134P
FU-234 -A1/6	52-234 -30/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 215P
FU-234 -A1/7	52-234 -32/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 135P
FU-234 -A1/8	52-234 -34/A1	SIS HT TR DIST PNL A1	HT TRACE CKT 216P

(RR)

52-235 -2/17	FU-235 -2/F17	120VAC VIT PWR BD 1-II	PAS CONTMT AIR ISLN VLVS
52-235 -1/17	FU-235 -1/F17	120VAC VIT PWR BD 1-I	PAS HOT LEG 1 ISLN VLVS
52-235 -4/36	FU-235 -4/F36	120VAC VIT PWR BD 1-IV	EGTS SHIELD BLDG VENT FL
52-235 -3/2	FU-99 -R58/M19,M20	120VAC VIT PWR BD1-III	GLYCOL EXP TK
FU-99 -R55/L15,L16	52-235 -2/44	AUX RLY PNL 1-R-55	SIS ACCUM TK 4 FL ISLN VLV
FU-99 -R55/L17,L18	52-235 -1/43	AUX RLY PNL 1-R-55	SIS ACCUM TK 1 FL ISLN VLV
FU-77 -L2/LF2(+)	FU-77 -L2/LF2(-)	PNL-77-L2	REAC CLNT DR TK PMP CNTL
FU-77 -L2/LF17(+)	FU-77 -L2/LF17(-)	PNL-77-L2	REAC CLNT DR TK LVL CNLT VLV
FU-275 -R76/I1,I2	52-235 -12/III	AUX RLY PNL 1-R-76	INCORE INST A RM A/C AIR FL
FU-275 -R76/I23,I24	52-235 -5/IV	AUX RLY PNL 1-R-76	INCORE INST B RM A/C AIR FL

(SS)

8. <u>48VDC BDS</u>

Fuse 1	Fuse 1	Communication BD,	Loud SPKR
2	1	BAY 39, PNLM	
3	1		
4	1		
5	1		
8	1		
9	1		
10	1		
11	1		
12	1		
13	1		
3	1	Communication BD,	
4	1	BAY 38, PNLM	
5	1		
6	1		
7	1		
8	1		
9	1		
10	1		
13	1		
14	1		

Technical Specification Tables 3.8-2 and 3.8-3

Thermal Overload Bypass Devices

NRC approved a change to Table 3.8-2 which would make Technical Specification 3.8.4.2 agree with revision 5 to the Standard Technical Specifications (NUREG-0452); however, several errors were made when the changes were incorporated into the final draft of the Watts Bar Technical Specifications. Attached are marked-up pages to correct these errors.

Also, the titles for Tables 3.8-2 and 3.8-3 should be changed to accurately describe what is listed in the tables. Since it is the thermal overload which is bypassed, the word "BYPASS" should be removed from the titles for both Tables 3.8-2 and 3.8-3.

To prevent confusion, the column "BYPASS DEVICE" should be removed from Table 3.8-3. The five valves listed in Table 3.8-3 have bypass devices; however, they receive a unit 2 accident signal. Therefore, for the purposes of unit 1 technical specifications, these valves have been listed as "NOT BYPASSED UNDER ACCIDENT CONDITIONS." The inclusion of the "BYPASS DEVICE" column adds no useful information to Table 3.8-3 and would possibly cause Table 3.8-3 to be confused with Table 3.8-2.

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TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD ~~BYPASS~~ DEVICES WHICH ARE BYPASSED UNDER ACCIDENT CONDITIONS

VALVE NO.	FUNCTION	BYPASS DEVICE
1-FCV-62-63	Isolation for Seal Water Filter	Yes
1-FCV-62-138	Safe Shutdown Redundancy (CVCS)	Yes
1-FCV-62-98	ECCS Operation	Yes
1-FCV-62-99	ECCS Operation	Yes
1-FCV-62-90	ECCS Operation	Yes
1-FCV-62-91	ECCS Operation	Yes
1-FCV-62-61	Cont. Isolation	Yes
1-LCV-62-132	ECCS Operation	Yes
1-LCV-62-133	ECCS Operation	Yes
1-LCV-62-135	ECCS Operation	Yes
1-LCV-62-136	ECCS Operation	Yes
1-FCV-74-1	Open for Normal Plant Cooldown	Yes
1-FCV-74-2	Open for Normal Plant Cooldown	Yes
1-FCV-74-3	ECCS Operation	Yes
1-FCV-74-21	ECCS Operation	Yes
1-FCV-74-12	RHR Pump, Mini-flow Protects Pump	Yes
1-FCV-74-24	RHR Pump, Mini-flow Protects Pump	Yes
1-FCV-74-33	ECCS Operation	Yes
1-FCV-74-35	ECCS Operation	Yes
1-FCV-63-7	ECCS Operation	Yes
1-FCV-63-6	ECCS Operation	Yes
1-FCV-63-156	ECCS Flow Path	Yes
1-FCV-63-157	ECCS Flow Path	Yes
1-FCV-63-39	BIT Injection	Yes
1-FCV-63-40	BIT Injection	Yes
1-FCV-63-25	BIT Injection	Yes
1-FCV-63-26	BIT Injection	Yes
1-FCV-63-118	RCS Pressure Boundary	Yes
1-FCV-63-98	RCS Pressure Boundary	Yes
1-FCV-63-80	RCS Pressure Boundary	Yes
1-FCV-63-67	RCS Pressure Boundary	Yes
1-FCV-63-1	ECCS Operation	Yes
1-FCV-63-72	ECCS Flow Path from Cont. Sump	Yes
1-FCV-63-73	ECCS Flow Path from Cont. Sump	Yes
1-FCV-63-8	ECCS Flow Path	Yes
1-FCV-63-11	ECCS Flow Path	Yes
1-FCV-63-93	ECCS Cooldown Flow Path	Yes
1-FCV-63-94	ECCS Cooldown Flow Path	Yes
1-FCV-63-172	ECCS Flow Path	Yes
1-FCV-63-5	ECCS Flow Path	Yes
1-FCV-63-47	Train Isolation	Yes
1-FCV-63-48	Train Isolation	Yes
1-FCV-63-4	SI Pump Mini-flow	Yes
1-FCV-63-175	SI Pump Mini-flow	Yes

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD ~~BYPASS~~ DEVICES WHICH ARE BYPASSED UNDER ACCIDENT CONDITIONS

VALVE NO.	FUNCTION	BYPASS DEVICE
1-FCV-63-177	SIS Pump Inlet to CVCS	Yes
1-FCV-63-3	SI Pump Mini-Flow	Yes
1-FCV-63-152	ECCS Recirc	Yes
1-FCV-63-153	ECCS Recirc	Yes
1-FCV-63-22	ECCS Recirc	Yes
1-FCV-3-33	Quick Closing Isolation	Yes
1-FCV-3-47	Quick Closing Isolation	Yes
1-FCV-3-87	Quick Closing Isolation	Yes
1-FCV-3-100	Quick Closing Isolation	Yes
1-FCV-1-15	Stm Supply to Aux FWP turbine	Yes
1-FCV-1-16	Stm Supply to Aux FWP turbine	Yes
1-FCV-3-179A	ERCW Sys Supply to Pump	Yes
1-FCV-3-179B	ERCW Sys Sypply to Pump	Yes
1-FCV-3-136A	ERCW Sys Supply to Pump	Yes
1-FCV-3-136B	ERCW Sys Supply to Pump	Yes
1-FCV-3-116A	ERCW Sys Supply to Pump	Yes
1-FCV-3-116B	ERCW Sys Supply to Pump	Yes
1-FCV-3-126A	ERCW Sys Supply to Pump	Yes
1-FCV-3-126B	ERCW Sys Supply to Pump	Yes
1-FCV-70-133	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-70-139	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-70-4	Isolation for Non-Essential Loads	Yes
1-FCV-70-143	Isolation for Excess Letdown Ht Xchngr	Yes
1-FCV-70-92	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-70-90	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-70-87	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-70-89	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-70-140	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-70-134	Isolation for RCP Oil Coolers & Therm B	Yes
1-FCV-67-67	DG Ht Ex	Yes
1-FCV-67-66	DG Ht Ex	Yes
1-FCV-67-123	CS Ht Ex Supply	Yes
1-FCV-67-125	CS Ht Ex Supply	Yes
1-FCV-67-124	CS Ht Ex Discharge	Yes
1-FCV-67-126	CS Ht Ex Discharge	Yes
0-FCV-67-151	CCWS Ht Ex Throttling	Yes
1-FCV-67-146	CCWS Ht Ex Throttling	Yes
1-FCV-67-223	Isolation of 1B/2A HDR's	Yes
1-FCV-67-83	Cont. Isol. Lower	Yes
1-FCV-67-88	Cont. Isol. Lower	Yes
1-FCV-67-87	Cont. Isol. Lower	Yes
1-FCV-1-51	AFPT Trip and Throttle Valve	Yes
1-FCV-67-68	DG Ht Ex	Yes
1 ~~2~~-FCV-67-65	DG Ht Ex	Yes

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD ~~BYPASS~~ DEVICES WHICH ARE BYPASSED UNDER ACCIDENT CONDITIONS

VALVE NO.	FUNCTION	BYPASS DEVICE
1-FCV-67-95	Cont. Isol. Lower	Yes
1-FCV-67-96	Cont. Isol. Lower	Yes
1-FCV-67-91	Cont. Isol. Lower	Yes
1-FCV-67-103	Cont. Isol. Lower	Yes
1-FCV-67-104	Cont. Isol. Lower	Yes
1-FCV-67-99	Cont. Isol. Lower	Yes
1-FCV-67-111	Cont. Isol. Lower	Yes
1-FCV-67-112	Cont. Isol. Lower	Yes
1-FCV-67-107	Cont. Isol. Lower	Yes
1-FCV-67-130	Cont. Isol. Upper	Yes
1-FCV-67-131	Cont. Isol. Upper	Yes
1-FCV-67-295	Cont. Isol. Upper	Yes
1-FCV-67-134	Cont. Isol. Upper	Yes
1-FCV-67-296	Cont. Isol. Upper	Yes
1-FCV-67-133	Cont. Isol. Upper	Yes
1-FCV-67-139	Cont. Isol. Upper	Yes
1-FCV-67-297	Cont. Isol. Upper	Yes
1-FCV-67-138	Cont. Isol. Upper	Yes
1-FCV-67-142	Cont. Isol. Upper	Yes
1-FCV-67-298	Cont. Isol. Upper	Yes
1-FCV-67-141	Cont. Isol. Upper	Yes
1-FCV-72-21	Cont. Spray Pump Suction	Yes
1-FCV-72-22	Cont. Spray Pump Suction	Yes
1-FCV-72-2	Cont. Spray Isol.	Yes
1-FCV-72-39	Cont. Spray Isol.	Yes
1-FCV-72-40	RHR Cont. Spray Isol.	Yes
1-FCV-72-41	RHR Cont. Spray Isol.	Yes
1-FCV-72-44	Cont. Sump to Hdr A - Cont. Spray	Yes
1-FCV-72-45	Cont. Sump to Hdr B - Cont. Spray	Yes
1-FCV-26-240	Cont. Isol.	Yes
1-FCV-26-241	Annulus Isol.	Yes
1-FCV-26-242	Annulus Isol.	Yes
1-FCV-26-243	RCP Cont. Spray Isol.	Yes
1-FCV-26-244	Annulus Isol.	Yes
1-FCV-26-245	Annulus Isol.	Yes
1-FCV-68-332	RCS PRZR Rel.	Yes
1-FCV-68-333	RCS PRZR Rel.	Yes
1-FCV-70-153	RHR Ht Ex B-B Outlet	Yes
1-FCV-70-156	RHR Ht Ex A-A Outlet	Yes
1-FCV-70-207	Cont. Demin. Waste Evap. Bldg. Supply	Yes

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TABLE 3.8-3

MOTOR-OPERATED VALVES THERMAL OVERLOAD ~~BYPASS~~ DEVICES WHICH ARE NOT BYPASSED UNDER ACCIDENT CONDITIONS

VALVE NO.	FUNCTION	~~BYPASS DEVICE~~
2-FCV-67-66	DG Ht Ex	~~Yes~~
2-FCV-67-67	DG Ht Ex	~~Yes~~
0-FCV-67-152	CCWS Ht Ex Throttling	~~Yes~~
2-FCV-67-65	DG Ht Ex	~~Yes~~
2-FCV-67-68	DG Ht Ex	~~Yes~~

LCO 3.9.6.b AND S.R 4.9.6.2

The control rod drive shafts supplied by Westinghouse for Watts Bar Nuclear Plant are of a different design from those normally supplied to other Westinghouse plants. The drive shaft and control rod assembly combined will weigh approximately 977 pounds; therefore, to provide sufficient margin an auxilliary hoist with a minimum capacity of 1200 pounds should be referenced in this tech. spec. Also, the load indicator should prevent lifting loads in excess of 1190 pounds.

REFUELING OPERATIONS

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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~~ 1200 pounds, and

2) A load indicator which shall be used to prevent lifting loads in excess of ~~690~~ 1190 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 before the crane load exceeds 3400 pounds.

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~~ 1200 pounds.

REACTOR BUILDING PURGE VENTILATION SYSTEM

SURVEILLANCE REQUIREMENT 4.9.13

The reactor building purge ventilation system flow rate should be revised to be 14,000 cfm for both trains. The technical specifications previously listed the system flow rate as 14,000 cfm. The technical specifications were changed as a result of the test values obtained during the preoperational test. During that test train A delivered 10,700 cfm and train B delivered 12,700 cfm. However, these flow rates were obtained with both trains running simultaneously. In the future Watts Bar intends to run the surveillance test on only one train at a time. In this configuration, each train can deliver 14,000 cfm. This has been recently verified by testing. The technical specifications should be revised to list the system flow rate as 14,000 cfm. The technical specifications should be written to reflect the expected test conditions.

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3/4.9.13 REACTOR BUILDING PURGE VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.13 The Reactor Building Purge Ventilation Systems shall be OPERABLE.

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

ACTION:

a. With one Reactor Building Purge Ventilation System inoperable, CORE ALTERATIONS or movement of irradiated fuel within the containment may proceed provided the OPERABLE Reactor Building Purge Ventilation System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Reactor Building Purge Ventilation System OPERABLE, suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel within the containment until at least one Reactor Building Purge Ventilation System is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 The above required Reactor Building Purge Ventilation Systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

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

   1) Verifying that the system satisfies the in-place penetration and bypass leakage acceptance criteria of less than 1% and uses the test procedure guidance of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is equal to 14,000 cfm ~~rated flow~~ ±10% ~~(train A rated flow is 10,300 cfm and train B rated flow is 12,700 cfm)~~.

EXPLOSIVE GAS MIXTURE

Surveillance Requirement 4.11.2.5

The surveillance requirement was revised at the request of B. Perch (NRC) to correct deficiencies.

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT

SAFETY EVALUATION REPORT CHANGES
REQUIRED FOR CERTIFICATION

RCS PRESSURE ISOLATION VALVE LEAK RATES (SER 3.9.6)

Section 3.9.6 of the SER requires the technical specifications to include limits for RCS Pressure Isolation Valve (PIV) leakage. TVA submitted a proposed technical specification which is consistant with the Watts Bar design and NRC concerns in a February 8, 1985, letter from J. A. Domer to E. Adensam. It has not been approved pending CRGR review.

LOCKED CLOSED RHR BYPASS LINE VALVES (SER 5.4.3)

SER section 5.4.3 requires that the RHR bypass valves "be normally locked closed at all times with the power removed from the operators. Strict administrative keylock control must be demonstrated." The same SER section states that the "alternate path can be used if one of the normal isolation valves cannot be opened." TVA believes these statements to be contradictory for the Watts Bar design. It is our position that the valves should be normally closed with the power removed at the breaker.

At Watts Bar, the RHR isolation valves and the isolation bypass valves are located inside containment. Entry into this area of containment is controlled, and special work permits and radiation surveys are required for entry. Since the valves are not readily accessible and it may be necessary to open them quickly, it is not practical to lock them in the closed position.

This position was transmitted to NRC in a September 15, 1982 letter from L. M. Mills to E. Adensam, and in a January 25, 1983 letter from D. S. Kammer to E. Adensam.

This section of the SER also appears to be in conflict with section 7.6.2. Section 7.6.2 accurately reflects the method TVA plans to use to remove power and control the position of the valve.

ICE BED SPECIFICATIONS (SER 6.1.3, 6.5.4)

SER Section 6.1.3 describes the ice bed as having a sodium tetraborate concentration equivalent to 2000 ppm boron. TVA's criteria is specified as 1900$\pm$ 100 ppm boron. The sump chemistry calculations for Watts Bar are based on a concentration of 1800 ppm boron. The Watts Bar technical specification (3.6.5.1.a) require a minimum concentration of 1800 ppm boron. It is TVA's position that Section 6.1.3 of the SER should be updated to reflect a minimum boron concentration of 1800 ppm for the ice bed. This position was transmitted to NRC in a letter from D. S. Kammer to E. Adensam dated January 25, 1983.

SER Section 6.5.4 states that the technical specifications require a minimum ice bed pH of 8.5. The Watts Bar technical specifications (3.6.5.1.a) require the ice bed to have a pH of 9.0 to 9.5. It is TVAs position that the SER should be updated to reflect a range of values for the ice bed pH, since an excessively high pH can be detrimental to equipment, also.

ANNULUS BYPASS LEAKAGE (SER 6.2.6, 15.4.1)

Section 6.2.6 of the SER states that annulus bypass leakage is to be limited to 10% by technical specifications. It appears that Table 15-2 in SER section 15.4.1 assumes an annulus bypass leakage of 0%. Amendment 48 of the Watts Barr FSAR contained information supporting annulus bypass leakage of 25%. It is TVA's position that the SER should be updated to reflect the annulus bypass leakage value of 25% in the FSAR and the Watts Bar technical specifications. This position was transmitted to NRC by letter dated January 25, 1983 from D. S. Kammer to E. Adensam, and by letter dated June 24, 1982 from L. M. Mills to E. Adensam.

MANUAL RESET OF SAFETY INJECTION SIGNAL (SER 6.3.2)

SER section 6.3.2 requires that emergency procedures be established which preclude manual reset of the safety injuection signal for at least 10 minutes following the initiation of the safety injection signal. The WOG revised emergency procedure guidlines used by Watts Bar utilize specific safety injection termination criteria. These criteria ensure that the reactor is in a stable and safe condition before safety injection is terminated. TVA's response to FSAR question 40.70 was revised in Amendment 55 to reflect this position. The SER should be updated to reflect the safety injection termination procedures at Watts Bar.

ECCS VALVES WITH POWER REMOVED (SER 6.3.2, 7.6.4, 8.3.1.8)

SER Section 6.3.2 lists several Emergency Core Cooling System (ECCS) valves which NRC states must have power removed during normal plant operation to prevent inadvertent operation. NRC is concerned that inadvertent operation of these valves will prevent the valve's associated system from performing its safety function. A discussion of each of these valves is provided below:

1. Cold leg accumulator isolation valves (FCV-63-67, -80, -98, -118)

   These valves are open with power removed during power operations. Surveillance requirements 4.5.1.1.1.a.2 and 4.5.1.1.1.c verify that the valves are opened and that the power breakers are tagged open.

2. Hot leg injection line valves (FCV-63-156, -157, -172)

   These valves are normally closed. They are verified closed as part of the startup procedure. Subsequent operator actions in the transfer to hot leg recirculation procedure (ES-1.3) requires that these valves be opened during the accident mitigation sequence. It is not prudent to require an operator to leave the control room to put power back on these valves. In addition, these valves utilize a modified control circuit which ensures that no single failure can energize the opening or closing coils of the valve operator. Protection against inadvertent operation of a valve by personnel is provided by hinged clear plastic covers over each respective valve operator handswitch in the main control room. These modified circuits are discussed in sections 7.6.4 and 8.3.1.8 of the SER.

3. Cross connect valves for RHR discharge and high pressure pump suction (FCV-63-6, -7, -8, and -11)

   These valves are normally closed. They are verified closed as part of the startup procedure. Subsequent operator actions in ES-1.2 (Transfer to Containment Sump) require that these valves be opened during the accident mitigation sequence. It is not prudent to require an operator to leave the control room to put power back on these valves. Valves FCV-63-8 and -11 utilize the modified control circuit and hand switch covers as described in SER sections 7.6.4 and 8.3.1.8. The single failure of FCV-63-6 or -7 will not impair the performance of the safety function of the system. Because of this, these valves do not utilize the modified control circuit or plastic hand switch covers.

4. Containment Sump to RHR Suction Valves (FCV-63-72, -73)

These valves are normally closed during power operation. They are verified closed during the startup procedure. The valves automatically open upon receipt of a low RWST level and a high containment sump level. This automatic actuation initiates the transfer from injection mode to recirculation mode (ES-1.2). These valves utilize the modified control circuit and plastic handswitch covers as described in SER Sections 7.6.4 and 8.3.1.8 to prevent inadvertent valve actuation. It is not prudent to remove power from these valves.

5. RHR Discharge Valves (FCV-63-93, -94)

These valves are normally open. They are verified open during the startup procedure. These valves must be closed in the transfer from cold leg recirculation to hot leg recirculation (ES-1.3). These valves utilize the plastic handswitch covers and the modified control circuit to prevent inadvertent actuation of the valves as described in SER sections 7.6.4 and 8.3.1.8. It is not prudent to remove power from these valves.

6. RWST to SI Pump Suction (FCV-63-5)

This valve is normally open during power operation. It is verified open during the startup procedure. This valve is closed during the switchover from injection mode to recirculation mode (ES-1.2). This valve must be closed to provide double isolation (paired with check valve 63-510) to prevent highly radioactive sump water from reaching the RWST. This valve utilizes the modified control circuit and clear plastic handswitch covers as described in SER sections 7.6.4 and 8.3.1.8 to prevent inadvertent actuation of the valve. It is not prudent to remove power from this valve.

7. RWST to RHR Pump Suction (FCV-63-1)

This valve is open with the power removed as required by surveillance requirement 4.5.2.a.

8. SI Pump Miniflow Valves (FCV-63-3-4, -175)

These valves are normally open. They are verified open during the startup procedure. These valves must be closed during the switchover from injection mode to recirculation mode (ES-1.2) to prevent pumping highly radioactive water to the RWST. FCV-63-3 utilizes the modified control circuit and the plastic handswitch covers as described in SER sections 7.6.4 and 8.3.1.8. The single failure of either FCV-63-4 or -175 will not impair the performance of the safety function of the system. It is not prudent to remove power from these valves.

9. SI Pump Cold Leg Injection Lines (FCV-63-22, -152, -153)

These valves are normally open. They are verified open during the startup procedure. Valve FCV-63-22 is open with power removed as required by Surveillance requirement 4.5.2.a. Valves FCV-63-152 and -153 are closed during the switchover from cold leg recirculation to hot leg recirculation (ES-1.3) to terminate cold leg recirculation. The single failure of either FCV-63-152 or -153 will not impair the performance of the safety function of the system. It is not prudent to remove power from these valves.

TVA's discussion of these valves was previously transmitted to NRC in letters dated September 15, 1982, from L. M. Mills to E. Adensam, and January 25, 1983, from D. S. Kammer to E. Adensam. These valves were also discussed in a meeting held September 7, 1984, between TVA Staff and NRC Reactor Systems Branch Staff in Bethesda, Maryland. At this meeting it was determined that TVA's discussion of these valves satisfied the NRC concerns. It is TVA's position that SER section 6.3.2 be updated to clarify resolution of the NRC concerns. The design of the control circuits to the valves listed in section 7.6.6 of the FSAR effectively removes power from the opening and closing coils. SER section 6.3.2 should be made consistent SER sections 7.6.4 and 8.3.1.8.

REACTOR BUILDING PURGE SYSTEM FILTER EFFICIENCIES (SER 6.5.1.3)

SER section 6.5.1.3 describes the Reactor Building Purge Ventilation System. This section describes the system filter efficiencies as 95% for elemental and organic iodine. TVA has revised its fuel handling accident analysis to allow relaxed acceptance criteria of the filters to less than 10% penetration for methyl iodide. TVA's revised radiological analysis was based on filter efficiencies of 90% for inorganic iodine and 30% for organic iodine. These efficiencies were acceptable to the responsible NRC reviewers during an August 6-7, 1984 TVA/NRC technical specification meeting. There numbers were documented in a August 23, 1984 letter from L. M. Mills to E. Adensam. TVA believes that 6.5.1.3 should be updated to reflect the current filter efficiencies.

Proposed Anticipatory Trip Modification (SER 7.8.4)

Section 7.8.4 of the SER states that Watts Bar has not proposed a change in the 10% rated thermal power interlock for reactor trip on turbine trip. TVA has submitted a change for Watts Bar to increase the reactor trip on turbine trip interlock setpoint to 50% rated thermal power. The analysis for this change was submitted to NRC as TVA's response to NUREG-0737, Item II.K.3.12. The error in the SER was transmitted to NRC by letters dated June 24, 1982, from L. M. Mills to E. Adensam, and January 25, 1983, from D. S. Kammer to E. Adensam.

FUSE RESISTANCE TESTING (SER 8.3.3.6)

SER section 8.3.3.6 states that a requirement for periodic measurement of fuse and terminal connection resistance will be included in the technical specifications.

Periodic resistance measurement is not practical for containment penetration conductor overcurrent protection fuses. Resistance verification is performed as one of the final steps in the manufacturing process, assurring proper construction and rating. Manufacturers do not publish this baseline data since construction changes are made based on design and material improvements. Because of this, no baseline data would be available if periodic resistance measurements were performed. Routine removal of fuses for testing is not prudent according to the manufacturer. Routine removal can result in damaging of the fuse holder and contact points. In the case of cable protecting fuses, the fuse would be physically destroyed when it was removed because of the crimped joint used to connect it. Fuse manufacturers have also stated that fuses do not deteriorate with service life. Service temperatures above the rated temperature, current surges, and unusual cycling conditions all reduce the fuse's service life, i.e., the fuse becomes more protective. Under no condition will a fuse become less protective during its service life.

Because of this, TVA will suspend the resistance measurement requirement until NRC completes its generic study. This information was transmitted to NRC in Amendment 55 of FSAR section 8.1.5.3.

FIRE PROTECTION SYSTEM (SER Section 9.5.1.2)

Page 9.27 of the original SER states that the raw water system is automatically isolated when the fire pumps start. This statement is somewhat misleading. To clarify this, we believe the statement should be modified as shown on the marked-up page of the SER.

position and Technical Specification surveillance is placed upon supervision of valve position to ensure proper system alignment. The yard fire main loop is cross-tied between units. The fire protection headers are pressurized through an interconnection with the raw water system, with the pressure being maintained by two 10,000-gal raw water tanks on the auxiliary building roof. The raw water ~~system is~~ automatically isolated when the fire pumps start.
tanks are

The diesel generator building has a single feed from the underground fire main into the building feeding a preaction sprinkler system and manual hose station. A single active failure or crack in a moderate-energy line can impair both the primary and backup fire suppression systems. By letter dated August 28, 1981, the applicant agreed to provide an additional fire water feed for the diesel generator building so that the primary or secondary suppression is assured.

Based on its review, the staff concludes that the fire water supply system meets the guidelines of Section C.2 of Appendix A to BTP 9.5-1 and, therefore, is acceptable.

<u>Sprinkler and Standpipe Systems</u>

Automatic sprinkler systems and hose station standpipe systems are separately connected to the yard main or to headers within buildings fed from each end of the building; therefore, a single failure cannot impair both sprinkler systems and hose stations. Fixed-water spray systems and sprinkler systems are designed according to the requirements of NFPA Standard No. 13, "Standard for Installation of Sprinkler Systems," and NFPA Standard No. 15, "Standard for Water Spray Fixed System." Hose stations are provided for the lower levels of the intake pumping station; however, these hose stations cannot reach the upper elevations. By letter dated August 28, 1981, the applicant agreed to modify the present standpipe system by extending it and adding an additional interior manual hose station. Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety-related area in the plant. The system is designed according to the requirements of NFPA Standard No. 14, "Standpipe and Hose System for Sizing, Spacing, and Pipe Support Requirements." Pipe and pipe hangers of the fire protection system located in seismic Category I structures are designed for seismic requirements to ensure the integrity of other essential equipment in the same area.

Valves in the fire protection system are not electrically supervised; however, all valves whose misalignment would prevent proper operation of the system will be mechanically locked in their normal position. Technical Specifications surveillance is placed on supervision of valve position to ensure proper system alignment.

Areas that are equipped with water suppression systems are

<u>Control Building</u>

<u>Elevation 755 ft</u>

(1) mechanical equipment room
(2) janitor's closet
(3) corridor
(4) kitchen

CO_2 Fire Protection Areas (SER 9.5.1.2)

SER Section 9.5.1.2 lists the areas which are protected by low-pressure, total-flooding carbon dioxide (CO_2) systems. The CO_2 total-flood system has been removed from the cable spreading room. This change was transmitted to NRC by letter dated July 27, 1983, from D. S. Kammer to E. Adensam.

LOCKED FIRE PROTECTION VALVES (SER 9.5.1.2)

SER section 9.5.1.2 states that "all valves whose misalignment would prevent proper operation of the system will be mechanicaly locked in their normal position." It is TVA's position that this statement should be clarified to indicate that some valves are electrically supervised and only valves which are not electrically supervised will be mechanically locked in position.

TURBINE OVERSPEED PROTECTION (SER 10.2.1)

TVA has developed a technical specification in conjunction with NRC. This technical specification and justification were transmitted to NRC under seperate cover.

TURBINE BYPASS VALVE TESTING (SER 10.4.4, 16 ITEM 29)

Section 10.4.4 of the SER states that NRC will require a technical specification for the stroking of the turbine bypass valves on a periodic basis. The same section of the SER also states that the "turbine bypass system is not required for plant control following an accident and is not a safety-related system." It is TVA's position that these statements are contradictory.

The turbine bypass system at Watts Bar is not a safety-related system designed to mitigate any design basis events. The loss of offsite power will make the turbine bypass system inoperable.

The controls of the system were not designed with testability of the system at power planned. The quarterly testing proposed in the SER would require the testing to be performed at power. The 12 bypass valves would all have to be isolated for the stroke test or test jumpers and lifted leads would be required. Both test methods contradict NRC's safety concerns. TVA will not shut down Watts Bar on a quarterly basis to test these valves. TVA's position is that valve stroking on a refueling outage basis is sufficient and is consistant with intervals specified in ASME section XI for PORVs and safety valves. This position has been transmitted to NRC in a letter dated October 9, 1981 from L. M. Mills to E. Adensam, and a letter dated January 25, 1983 from D. S. Kammen to E. Adensam. The position was also transmitted as TVA's response to FSAR question 40.124. It is also TVA's position that technical specifications are not required because no significant safety hazard has been identified by NRC.

Gas Decay Tank Monitors (SER 11.3)

Section 11.3 of the SER requires that the reactor be shut down if a gas monitor is inoperable for more than 7 days. Standard Review Plan section 11.4 provides guidance for acceptable monitoring schemes; however, no mention of plant shutdown requirements is made. It is TVA's position that requiring a plant shutdown will lead to degassing at a time when one or both monitors are inoperable. We believe this to be a significant safety concern. TVA's position was transmitted to NRC by letter dated January 25, 1983, from D. S. Kammer to E. Adensam.

REACTOR TRIP ON HIGH STEAM GENERATOR LEVEL (SER 15.2)

SER section 15.2 lists that a high-steam generator water level will result in a reactor trip. Watts Bar FSAR section 15.2.10.2 states that a turbine trip will result from a steam generator high-high water level signal. The turbine trip will initiate a reactor trip only at power levels above 50%. It is TVA'S position that the SER should be updated to clarify this point. TVA's position was transmitted to NRC by letter dated March 5, 1982, from L. M. Mills to E. Adensam and by letter dated January 25, 1983, from D. S. Kammer to E. Adensam.

Bypass Reactor Trip Breaker Testing (SER 15.3.6)

SSER 3 section 15.3.6 states that the bypass breaker undervoltage trip attachment will be demonstrated operable at a refueling outage frequency. It is TVA's position that section 15.3.6 should be updated to clarify that the demonstration of operability does not include independent verification of the undervoltage and shunt coil trips. The WOG modification installed at Watts Bar is not designed for independent verification on the bypass breakers. The manual trip testing of the bypass breakers is believed to be sufficient since the bypass breakers are closed only during the testing of the reactor trip breakers.

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT

TECHNICAL SPECIFICAION CHANGES
THAT ARE AN ENHANCEMENT AND OPTIMIZATION
OF THE WATTS BAR TECHNICAL SPECIFICATIONS

SUMMARY OF PREVIOUSLY REQUESTED CHANGES

Page	Comment
2-6	Revised allowable values for the interlock channels were submitted to NRC by letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these technical specification changes.
2-6, 3/4 3-4, 3/4 3-12, 3/4 3-13	TVA requested changes to the P-13 interlock by letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these technical specification changes.
3/4 3-6, 3/4 3-7, 3/4 3-8, 3/4 3-11, 3/4 3-12, 3/4 3-13, 3/4 3-14, B 3/4 3-1	TVA requested changes to the reactor protection system consistent with the changes recommended by the Westinghouse Owners Group. NRC has approved the changes on a generic basis in a letter from C. O. Thomas to J. J. Sheppard dated February 21, 1985. TVA addressed the NRC conditions identified in their safety evaluation by letter from R. H. Shell to E. Adensam dated February 19, 1985. TVA continues to request these changes.
3/4 3-17, 3/4 3-20, 3/4 3-21, 3/4 3-25	TVA requested changes to the auxiliary feedwater and containment spray action statements to make them consistent with other portions of the technical specifications. This information was submitted to NRC by letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
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3/4 3-83, 3/4 3-84, 3/4 3-85, 3/4 3-87, 3/4 3-88	TVA requested changes to the mode requirements for various radiation monitors in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.

3/4 4-20, 3/4 4-21, 3/4 4-22, B3/4 4-5	TVA requested changes to the pressure isolation valve testing requirements in a letter from J. A. Domer to E. Adensam dated February 8, 1985. TVA continues to request these changes.
3/4 4-26, 3/4 4-27, B 3/4 4-7	TVA requested a change to the specific activity reporting requirements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 5-1, 3/4 5-3	TVA requested changes to the accumulator action statements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 5-2	TVA requested a deletion of the P-11 and safety injection testing for the accumulator valves in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 6-27, 3/4 6-28, B 3/4 6-4	TVA requested a reduction in ice weights in a letter from D. L. Lambert to E. Adensam dated January 10, 1985. Additional information was provided in a letter from J. W. Hufham to E. Adensam dated February 15, 1985. TVA continues to request these changes.
3/4 7-5	TVA requested a change to the auxiliary feedwater valve alignment requirements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
3/4 7-21 thru 3/4 7-26, B 3/4 7-4 thru B 3/4 7-6	TVA requested changes to the snubber test requirements in a letter from D. S. Kammer to E. Adensam dated June 19, 1984. Several meetings were held subsequent to that submittal. TVA has recently received some NRC data in a letter from E. Adensam to H. G. Parris dated March 6, 1985. TVA is reviewing the data and will pursue the snubber issue appropriately.
3/4 7-29	TVA requested a change to the fire pump test requirement in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
3/4 7-39	TVA requested a change to the area temperature monitoring action statements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change

3/4 8-1, 3/4 8-2, 3/4 8-8, 3/4 8-9	TVA requested changes to the diesel generator specification consistent with generic letter 84-15 in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 8-20, Table 3.8-1	TVA requested a change to delete Table 3.8-1 in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
3/4 11-18	TVA requested a change to surveillance requirement 4.11.2.6 in a letter from D.L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
B 3/4 4-15	TVA requested an addition to the bases for the low temperature overpressure protection system in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
B 3/4 11-3	TVA requested a change to the bases for gaseous dose rates in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. While not required for certification because it is in the bases, TVA believes the change should be made to accurately reflect how dose rates are calculated.

NEW REQUESTS

SURVEILLANCE REQUIREMENT 4.3.1.1.2.a

The monthly channel calibration of the power range is a single point comparison of incore to excore axial flux difference while operating above 15% RTP. This calibration is dependent upon core burnup and axial power distribution and not calendar time. It is also required to be performed while at power. Therefore, the monthly requirement should be changed to once per 31 EFPD.

A single point comparison of incore vs. excore axial flux difference is required to be performed once per month by SR 4.3.1.1.2.a. If a deviation of more than 3% is seen, a recalibration is required to be performed. This recalibration is accomplished by the performance of an Incore-Excore Cross Calibration. This test requires the plant to be placed in a xenon oscillation at a reduced power level. At BOL this oscillation is a converging one and is self-dampening but at EOL it has a potential to become diverging and difficult to suppress. Experience from Sequoyah Nuclear Plant has shown that the cross-calibration usually need only be performed at the beginning of each cycle and once more toward EOL as determined by the monthly check above. The reduced power level is costly in lost generation because the test requires the plant to be at a reduced power level for 48 hours or more. Therefore, the quarterly requirement should be changed to once per refueling.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1,2,3*,4*,5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), once per 31 EFPD ~~M~~(3, 4), ~~Q(4, 6),~~ R(4, 5~~)~~, 6)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	$1^{\#\#\#}$, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	$1^{\#\#\#}$, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	$2^{\#\#}$, 3, 4, 5
7. Overtemperature ΔT	S	R(12)	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High	S	R	M	N.A.	N.A.	1
12. Reactor Coolant Flow -Low-Single Loop	S	R	M	N.A.	N.A.	1

FINAL DRAFT

RADIOACTIVE GASEOUS WASTE MONITORING SAMPLING AND ANALYSIS PROGRAM

Table 4.11-2

Table notation (4) should be revised to require tritium grab samples on a daily basis only when spent fuel is present in the reactor or spent fuel pool. This will eliminate the need for daily tritium sampling during initial core loading. No reduction in the level of protection results from this change because the fuel has not been initiated. This change also makes note (4) consistent with note (5) with respect to applicability.

TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(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 unless (a) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased by more than a factor of 3, and (b) the lower containment noble gas activity monitor (RE-90-106 or RE-90-112) shows that the radioactivity has not increased by more than a factor of 3.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded whenever spent fuel is in the spent fuel pool or reactor.

(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 by more than a factor of 3; and (2) the noble gas monitor shows that the radio activity has not increased by 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 or equal to 1×10^{-6} μCi/gm and the discharge Radiation Monitor Setpoint is less than or equal to 1×10^{-6} μCi/ml above background.

Technical Specification Page B 3/4 3-1

Attached is a marked-up copy of technical specification page B 3/4 3-1. This change will clarify item 8.b.2 of Table 3.3-3.

ANNULUS BYPASS LEAKAGE (SER 6.2.6, 15.4.1)

Section 6.2.6 of the SER states that annulus bypass leakage is to be limited to 10% by technical specifications. It appears that Table 15-2 in SER section 15.4.1 assumes an annulus bypass leakage of 0%. Amendment 48 of the Watts Barr FSAR contained information supporting annulus bypass leakage of 25%. It is TVA's position that the SER should be updated to reflect the annulus bypass leakage value of 25% in the FSAR and the Watts Bar technical specifications. This position was transmitted to NRC by letter dated January 25, 1983 from D. S. Kammer to E. Adensam, and by letter dated June 24, 1982 from L. M. Mills to E. Adensam.

MANUAL RESET OF SAFETY INJECTION SIGNAL (SER 6.3.2)

SER section 6.3.2 requires that emergency procedures be established which preclude manual reset of the safety injuection signal for at least 10 minutes following the initiation of the safety injection signal. The WOG revised emergency procedure guidlines used by Watts Bar utilize specific safety injection termination criteria. These criteria ensure that the reactor is in a stable and safe condition before safety injection is terminated. TVA's response to FSAR question 40.70 was revised in Amendment 55 to reflect this position. The SER should be updated to reflect the safety injection termination procedures at Watts Bar.

ECCS VALVES WITH POWER REMOVED (SER 6.3.2, 7.6.4, 8.3.1.8)

SER Section 6.3.2 lists several Emergency Core Cooling System (ECCS) valves which NRC states must have power removed during normal plant operation to prevent inadvertent operation. NRC is concerned that inadvertent operation of these valves will prevent the valve's associated system from performing its safety function. A discussion of each of these valves is provided below:

1. Cold leg accumulator isolation valves (FCV-63-67, -80, -98, -118)

   These valves are open with power removed during power operations. Surveillance requirements 4.5.1.1.1.a.2 and 4.5.1.1.1.c verify that the valves are opened and that the power breakers are tagged open.

2. Hot leg injection line valves (FCV-63-156, -157, -172)

   These valves are normally closed. They are verified closed as part of the startup procedure. Subsequent operator actions in the transfer to hot leg recirculation procedure (ES-1.3) requires that these valves be opened during the accident mitigation sequence. It is not prudent to require an operator to leave the control room to put power back on these valves. In addition, these valves utilize a modified control circuit which ensures that no single failure can energize the opening or closing coils of the valve operator. Protection against inadvertent operation of a valve by personnel is provided by hinged clear plastic covers over each respective valve operator handswitch in the main control room. These modified circuits are discussed in sections 7.6.4 and 8.3.1.8 of the SER.

3. Cross connect valves for RHR discharge and high pressure pump suction (FCV-63-6, -7, -8, and -11)

   These valves are normally closed. They are verified closed as part of the startup procedure. Subsequent operator actions in ES-1.2 (Transfer to Containment Sump) require that these valves be opened during the accident mitigation sequence. It is not prudent to require an operator to leave the control room to put power back on these valves. Valves FCV-63-8 and -11 utilize the modified control circuit and hand switch covers as described in SER sections 7.6.4 and 8.3.1.8. The single failure of FCV-63-6 or -7 will not impair the performance of the safety function of the system. Because of this, these valves do not utilize the modified control circuit or plastic hand switch covers.

4. Containment Sump to RHR Suction Valves (FCV-63-72, -73)

These valves are normally closed during power operation. They are verified closed during the startup procedure. The valves automatically open upon receipt of a low RWST level and a high containment sump level. This automatic actuation initiates the transfer from injection mode to recirculation mode (ES-1.2). These valves utilize the modified control circuit and plastic handswitch covers as described in SER Sections 7.6.4 and 8.3.1.8 to prevent inadvertent valve actuation. It is not prudent to remove power from these valves.

5. RHR Discharge Valves (FCV-63-93, -94)

These valves are normally open. They are verified open during the startup procedure. These valves must be closed in the transfer from cold leg recirculation to hot leg recirculation (ES-1.3). These valves utilize the plastic handswitch covers and the modified control circuit to prevent inadvertent actuation of the valves as described in SER sections 7.6.4 and 8.3.1.8. It is not prudent to remove power from these valves.

6. RWST to SI Pump Suction (FCV-63-5)

This valve is normally open during power operation. It is verified open during the startup procedure. This valve is closed during the switchover from injection mode to recirculation mode (ES-1.2). This valve must be closed to provide double isolation (paired with check valve 63-510) to prevent highly radioactive sump water from reaching the RWST. This valve utilizes the modified control circuit and clear plastic handswitch covers as described in SER sections 7.6.4 and 8.3.1.8 to prevent inadvertent actuation of the valve. It is not prudent to remove power from this valve.

7. RWST to RHR Pump Suction (FCV-63-1)

This valve is open with the power removed as required by surveillance requirement 4.5.2.a.

8. SI Pump Miniflow Valves (FCV-63-3-4, -175)

These valves are normally open. They are verified open during the startup procedure. These valves must be closed during the switchover from injection mode to recirculation mode (ES-1.2) to prevent pumping highly radioactive water to the RWST. FCV-63-3 utilizes the modified control circuit and the plastic handswitch covers as described in SER sections 7.6.4 and 8.3.1.8. The single failure of either FCV-63-4 or -175 will not impair the performance of the safety function of the system. It is not prudent to remove power from these valves.

9. SI Pump Cold Leg Injection Lines (FCV-63-22, -152, -153)

These valves are normally open. They are verified open during the startup procedure. Valve FCV-63-22 is open with power removed as required by Surveillance requirement 4.5.2.a. Valves FCV-63-152 and -153 are closed during the switchover from cold leg recirculation to hot leg recirculation (ES-1.3) to terminate cold leg recirculation. The single failure of either FCV-63-152 or -153 will not impair the performance of the safety function of the system. It is not prudent to remove power from these valves.

TVA's discussion of these valves was previously transmitted to NRC in letters dated September 15, 1982, from L. M. Mills to E. Adensam, and January 25, 1983, from D. S. Kammer to E. Adensam. These valves were also discussed in a meeting held September 7, 1984, between TVA Staff and NRC Reactor Systems Branch Staff in Bethesda, Maryland. At this meeting it was determined that TVA's discussion of these valves satisfied the NRC concerns. It is TVA's position that SER section 6.3.2 be updated to clarify resolution of the NRC concerns. The design of the control circuits to the valves listed in section 7.6.6 of the FSAR effectively removes power from the opening and closing coils. SER section 6.3.2 should be made consistent SER sections 7.6.4 and 8.3.1.8.

REACTOR BUILDING PURGE SYSTEM FILTER EFFICIENCIES (SER 6.5.1.3)

SER section 6.5.1.3 describes the Reactor Building Purge Ventilation System. This section describes the system filter efficiencies as 95% for elemental and organic iodine. TVA has revised its fuel handling accident analysis to allow relaxed acceptance criteria of the filters to less than 10% penetration for methyl iodide. TVA's revised radiological analysis was based on filter efficiencies of 90% for inorganic iodine and 30% for organic iodine. These efficiencies were acceptable to the responsible NRC reviewers during an August 6-7, 1984 TVA/NRC technical specification meeting. There numbers were documented in a August 23, 1984 letter from L. M. Mills to E. Adensam. TVA believes that 6.5.1.3 should be updated to reflect the current filter efficiencies.

Proposed Anticipatory Trip Modification (SER 7.8.4)

Section 7.8.4 of the SER states that Watts Bar has not proposed a change in the 10% rated thermal power interlock for reactor trip on turbine trip. TVA has submitted a change for Watts Bar to increase the reactor trip on turbine trip interlock setpoint to 50% rated thermal power. The analysis for this change was submitted to NRC as TVA's response to NUREG-0737, Item II.K.3.12. The error in the SER was transmitted to NRC by letters dated June 24, 1982, from L. M. Mills to E. Adensam, and January 25, 1983, from D. S. Kammer to E. Adensam.

FUSE RESISTANCE TESTING (SER 8.3.3.6)

SER section 8.3.3.6 states that a requirement for periodic measurement of fuse and terminal connection resistance will be included in the technical specifications.

Periodic resistance measurement is not practical for containment penetration conductor overcurrent protection fuses. Resistance verification is performed as one of the final steps in the manufacturing process, assurring proper construction and rating. Manufacturers do not publish this baseline data since construction changes are made based on design and material improvements. Because of this, no baseline data would be available if periodic resistance measurements were performed. Routine removal of fuses for testing is not prudent according to the manufacturer. Routine removal can result in damaging of the fuse holder and contact points. In the case of cable protecting fuses, the fuse would be physically destroyed when it was removed because of the crimped joint used to connect it. Fuse manufacturers have also stated that fuses do not deteriorate with service life. Service temperatures above the rated temperature, current surges, and unusual cycling conditions all reduce the fuse's service life, i.e., the fuse becomes more protective. Under no condition will a fuse become less protective during its service life.

Because of this, TVA will suspend the resistance measurement requirement until NRC completes its generic study. This information was transmitted to NRC in Amendment 55 of FSAR section 8.1.5.3.

FIRE PROTECTION SYSTEM (SER Section 9.5.1.2)

Page 9.27 of the original SER states that the raw water system is automatically isolated when the fire pumps start. This statement is somewhat misleading. To clarify this, we believe the statement should be modified as shown on the marked-up page of the SER.

position and Technical Specification surveillance is placed upon supervision of valve position to ensure proper system alignment. The yard fire main loop is cross-tied between units. The fire protection headers are pressurized through an interconnection with the raw water system, with the pressure being maintained by two 10,000-gal raw water tanks on the auxiliary building roof. The raw water ~~system is~~ tanks are automatically isolated when the fire pumps start.

The diesel generator building has a single feed from the underground fire main into the building feeding a preaction sprinkler system and manual hose station. A single active failure or crack in a moderate-energy line can impair both the primary and backup fire suppression systems. By letter dated August 28, 1981, the applicant agreed to provide an additional fire water feed for the diesel generator building so that the primary or secondary suppression is assured.

Based on its review, the staff concludes that the fire water supply system meets the guidelines of Section C.2 of Appendix A to BTP 9.5-1 and, therefore, is acceptable.

Sprinkler and Standpipe Systems

Automatic sprinkler systems and hose station standpipe systems are separately connected to the yard main or to headers within buildings fed from each end of the building; therefore, a single failure cannot impair both sprinkler systems and hose stations. Fixed-water spray systems and sprinkler systems are designed according to the requirements of NFPA Standard No. 13, "Standard for Installation of Sprinkler Systems," and NFPA Standard No. 15, "Standard for Water Spray Fixed System." Hose stations are provided for the lower levels of the intake pumping station; however, these hose stations cannot reach the upper elevations. By letter dated August 28, 1981, the applicant agreed to modify the present standpipe system by extending it and adding an additional interior manual hose station. Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety-related area in the plant. The system is designed according to the requirements of NFPA Standard No. 14, "Standpipe and Hose System for Sizing, Spacing, and Pipe Support Requirements." Pipe and pipe hangers of the fire protection system located in seismic Category I structures are designed for seismic requirements to ensure the integrity of other essential equipment in the same area.

Valves in the fire protection system are not electrically supervised; however, all valves whose misalignment would prevent proper operation of the system will be mechanically locked in their normal position. Technical Specifications surveillance is placed on supervision of valve position to ensure proper system alignment.

Areas that are equipped with water suppression systems are

Control Building

Elevation 755 ft

(1) mechanical equipment room
(2) janitor's closet
(3) corridor
(4) kitchen

CO_2 Fire Protection Areas (SER 9.5.1.2)

SER Section 9.5.1.2 lists the areas which are protected by low-pressure, total-flooding carbon dioxide (CO_2) systems. The CO_2 total-flood system has been removed from the cable spreading room. This change was transmitted to NRC by letter dated July 27, 1983, from D. S. Kammer to E. Adensam.

LOCKED FIRE PROTECTION VALVES (SER 9.5.1.2)

SER section 9.5.1.2 states that "all valves whose misalignment would prevent proper operation of the system will be mechanicaly locked in their normal position." It is TVA's position that this statement should be clarified to indicate that some valves are electrically supervised and only valves which are not electrically supervised will be mechanically locked in position.

TURBINE OVERSPEED PROTECTION (SER 10.2.1)

TVA has developed a technical specification in conjunction with NRC. This technical specification and justification were transmitted to NRC under seperate cover.

TURBINE BYPASS VALVE TESTING (SER 10.4.4, 16 ITEM 29)

Section 10.4.4 of the SER states that NRC will require a technical specification for the stroking of the turbine bypass valves on a periodic basis. The same section of the SER also states that the "turbine bypass system is not required for plant control following an accident and is not a safety-related system." It is TVA's position that these statements are contradictory.

The turbine bypass system at Watts Bar is not a safety-related system designed to mitigate any design basis events. The loss of offsite power will make the turbine bypass system inoperable.

The controls of the system were not designed with testability of the system at power planned. The quarterly testing proposed in the SER would require the testing to be performed at power. The 12 bypass valves would all have to be isolated for the stroke test or test jumpers and lifted leads would be required. Both test methods contradict NRC's safety concerns. TVA will not shut down Watts Bar on a quarterly basis to test these valves. TVA's position is that valve stroking on a refueling outage basis is sufficient and is consistant with intervals specified in ASME section XI for PORVs and safety valves. This position has been transmitted to NRC in a letter dated October 9, 1981 from L. M. Mills to E. Adensam, and a letter dated January 25, 1983 from D. S. Kammen to E. Adensam. The position was also transmitted as TVA's response to FSAR question 40.124. It is also TVA's position that technical specifications are not required because no significant safety hazard has been identified by NRC.

Gas Decay Tank Monitors (SER 11.3)

Section 11.3 of the SER requires that the reactor be shut down if a gas monitor is inoperable for more than 7 days. Standard Review Plan section 11.4 provides guidance for acceptable monitoring schemes; however, no mention of plant shutdown requirements is made. It is TVA's position that requiring a plant shutdown will lead to degassing at a time when one or both monitors are inoperable. We believe this to be a significant safety concern. TVA's position was transmitted to NRC by letter dated January 25, 1983, from D. S. Kammer to E. Adensam.

REACTOR TRIP ON HIGH STEAM GENERATOR LEVEL (SER 15.2)

SER section 15.2 lists that a high-steam generator water level will result in a reactor trip. Watts Bar FSAR section 15.2.10.2 states that a turbine trip will result from a steam generator high-high water level signal. The turbine trip will initiate a reactor trip only at power levels above 50%. It is TVA'S position that the SER should be updated to clarify this point. TVA's position was transmitted to NRC by letter dated March 5, 1982, from L. M. Mills to E. Adensam and by letter dated January 25, 1983, from D. S. Kammer to E. Adensam.

Bypass Reactor Trip Breaker Testing (SER 15.3.6)

SSER 3 section 15.3.6 states that the bypass breaker undervoltage trip attachment will be demonstrated operable at a refueling outage frequency. It is TVA's position that section 15.3.6 should be updated to clarify that the demonstration of operability does not include independent verification of the undervoltage and shunt coil trips. The WOG modification installed at Watts Bar is not designed for independent verification on the bypass breakers. The manual trip testing of the bypass breakers is believed to be sufficient since the bypass breakers are closed only during the testing of the reactor trip breakers.

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT

TECHNICAL SPECIFICAION CHANGES
THAT ARE AN ENHANCEMENT AND OPTIMIZATION
OF THE WATTS BAR TECHNICAL SPECIFICATIONS

<u>SUMMARY OF PREVIOUSLY REQUESTED CHANGES</u>

Page	Comment
2-6	Revised allowable values for the interlock channels were submitted to NRC by letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these technical specification changes.
2-6, 3/4 3-4, 3/4 3-12, 3/4 3-13	TVA requested changes to the P-13 interlock by letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these technical specification changes.
3/4 3-6, 3/4 3-7, 3/4 3-8, 3/4 3-11, 3/4 3-12, 3/4 3-13, 3/4 3-14, B 3/4 3-1	TVA requested changes to the reactor protection system consistent with the changes recommended by the Westinghouse Owners Group. NRC has approved the changes on a generic basis in a letter from C. O. Thomas to J. J. Sheppard dated February 21, 1985. TVA addressed the NRC conditions identified in their safety evaluation by letter from R. H. Shell to E. Adensam dated February 19, 1985. TVA continues to request these changes.
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3/4 4-26, 3/4 4-27, B 3/4 4-7	TVA requested a change to the specific activity reporting requirements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 5-1, 3/4 5-3	TVA requested changes to the accumulator action statements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 5-2	TVA requested a deletion of the P-11 and safety injection testing for the accumulator valves in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 6-27, 3/4 6-28, B 3/4 6-4	TVA requested a reduction in ice weights in a letter from D. L. Lambert to E. Adensam dated January 10, 1985. Additional information was provided in a letter from J. W. Hufham to E. Adensam dated February 15, 1985. TVA continues to request these changes.
3/4 7-5	TVA requested a change to the auxiliary feedwater valve alignment requirements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
3/4 7-21 thru 3/4 7-26, B 3/4 7-4 thru B 3/4 7-6	TVA requested changes to the snubber test requirements in a letter from D. S. Kammer to E. Adensam dated June 19, 1984. Several meetings were held subsequent to that submittal. TVA has recently received some NRC data in a letter from E. Adensam to H. G. Parris dated March 6, 1985. TVA is reviewing the data and will pursue the snubber issue appropriately.
3/4 7-29	TVA requested a change to the fire pump test requirement in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
3/4 7-39	TVA requested a change to the area temperature monitoring action statements in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change

3/4 8-1, 3/4 8-2, 3/4 8-8, 3/4 8-9	TVA requested changes to the diesel generator specification consistent with generic letter 84-15 in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request these changes.
3/4 8-20, Table 3.8-1	TVA requested a change to delete Table 3.8-1 in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
3/4 11-18	TVA requested a change to surveillance requirement 4.11.2.6 in a letter from D.L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
B 3/4 4-15	TVA requested an addition to the bases for the low temperature overpressure protection system in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. TVA continues to request this change.
B 3/4 11-3	TVA requested a change to the bases for gaseous dose rates in a letter from D. L. Lambert to E. Adensam dated January 30, 1985. While not required for certification because it is in the bases, TVA believes the change should be made to accurately reflect how dose rates are calculated.

NEW REQUESTS

SURVEILLANCE REQUIREMENT 4.3.1.1.2.a

The monthly channel calibration of the power range is a single point comparison of incore to excore axial flux difference while operating above 15% RTP. This calibration is dependent upon core burnup and axial power distribution and not calendar time. It is also required to be performed while at power. Therefore, the monthly requirement should be changed to once per 31 EFPD.

A single point comparison of incore vs. excore axial flux difference is required to be performed once per month by SR 4.3.1.1.2.a. If a deviation of more than 3% is seen, a recalibration is required to be performed. This recalibration is accomplished by the performance of an Incore-Excore Cross Calibration. This test requires the plant to be placed in a xenon oscillation at a reduced power level. At BOL this oscillation is a converging one and is self-dampening but at EOL it has a potential to become diverging and difficult to suppress. Experience from Sequoyah Nuclear Plant has shown that the cross-calibration usually need only be performed at the beginning of each cycle and once more toward EOL as determined by the monthly check above. The reduced power level is costly in lost generation because the test requires the plant to be at a reduced power level for 48 hours or more. Therefore, the quarterly requirement should be changed to once per refueling.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1,2,3*,4*,5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), once per 31 EFPD ~~M~~(3, 4), ~~Q(4, 6);~~ R(4, 5~~)~~, 6)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(12)	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High	S	R	M	N.A.	N.A.	1
12. Reactor Coolant Flow -Low-Single Loop	S	R	M	N.A.	N.A.	1

FINAL DRAFT

RADIOACTIVE GASEOUS WASTE MONITORING SAMPLING AND ANALYSIS PROGRAM

Table 4.11-2

Table notation (4) should be revised to require tritium grab samples on a daily basis only when spent fuel is present in the reactor or spent fuel pool. This will eliminate the need for daily tritium sampling during initial core loading. No reduction in the level of protection results from this change because the fuel has not been initiated. This change also makes note (4) consistent with note (5) with respect to applicability.

FINAL DRAFT
FEB 15 1985

TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(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 unless (a) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased by more than a factor of 3, and (b) the lower containment noble gas activity monitor (RE-90-106 or RE-90-112) shows that the radioactivity has not increased by more than a factor of 3.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded whenever spent fuel is in the spent fuel pool or reactor.

(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 by more than a factor of 3; and (2) the noble gas monitor shows that the radio activity has not increased by 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 or equal to 1×10^{-6} μCi/gm and the discharge Radiation Monitor Setpoint is less than or equal to 1×10^{-6} μCi/ml above background.

Technical Specification Page B 3/4 3-1

Attached is a marked-up copy of technical specification page B 3/4 3-1. This change will clarify item 8.b.2 of Table 3.3-3.

3/4.3 INSTRUMENTATION

BASES

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

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

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

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) Phase A containment isolation, (6) Turbine trip, (7) auxiliary feedwater pumps start, (8) containment air return fans start, (9) essential raw cooling water pumps start and automatic valves position, (10) Control Room Isolation And Ventilation Systems start, and (11) component cooling water pumps start.

Diesel generator start from degraded voltage relays is accomplished after 300 seconds if SI is not present and after 10 seconds if SI is present through the undervoltage relays.

AUG 7 1984

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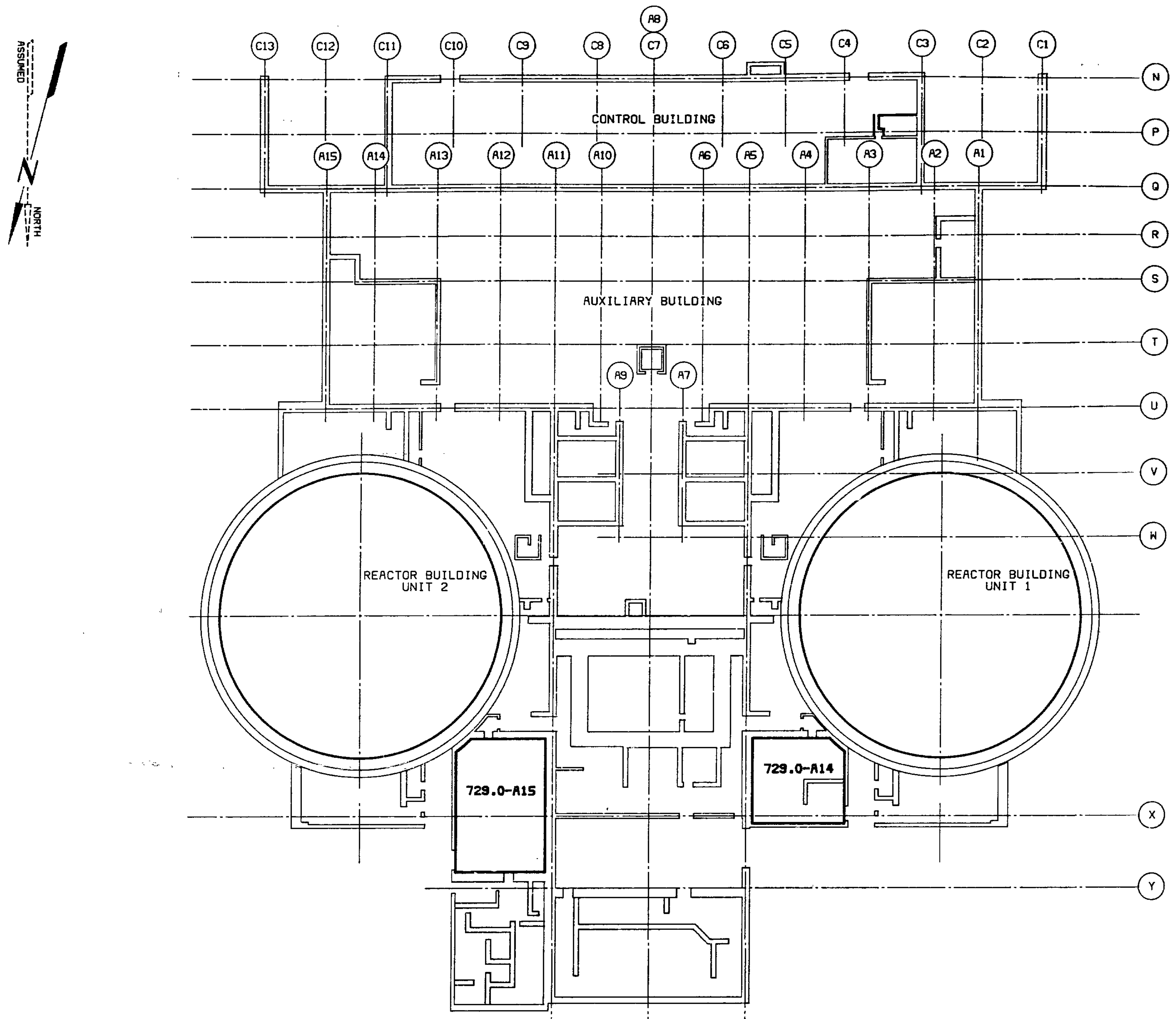
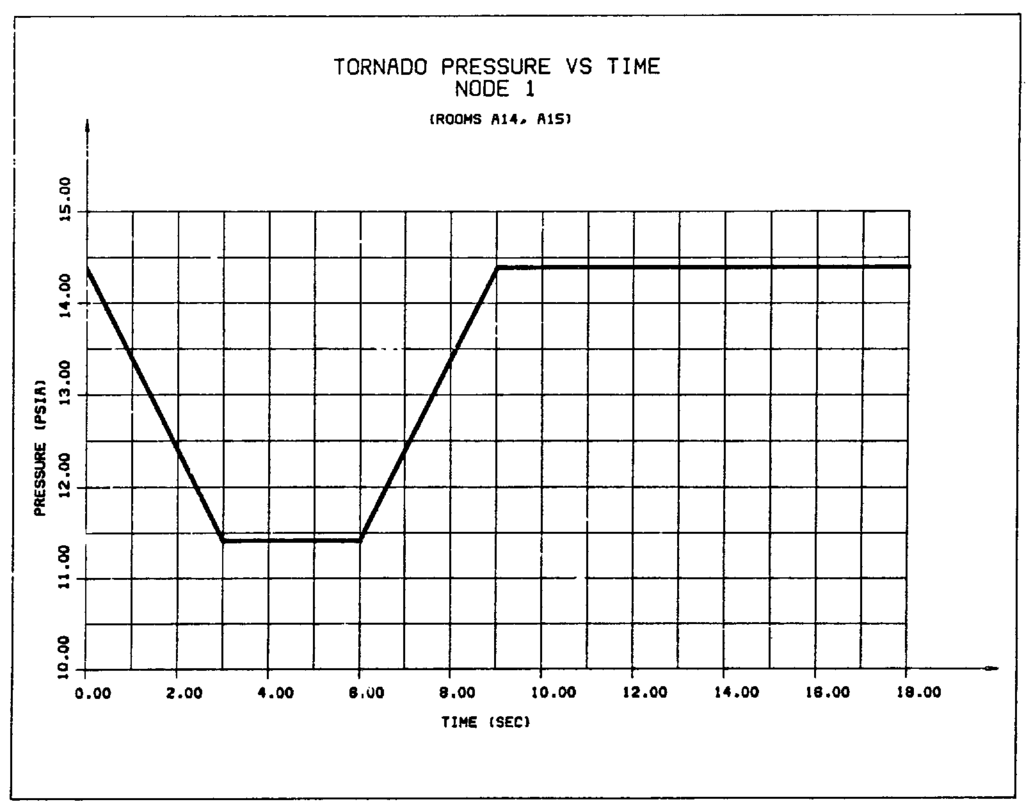


PLAN - EL 729.0 & 737.0

BUILDING LOCATION AND ROOM NUMBER	OPERATIONAL CONDITION (NOTE A)	TEMPERATURE (°F)	RELATIVE HUMIDITY (%) (NOTE G)	PRESSURE (PSIA) (NOTE B & H)	TOTAL 40 YEAR INTEGRATED DOSE (RADS) (NOTE C)	INTEGRATED ACCIDENT DOSE (RADS) (NOTES A & C)	AREA TYPE (NOTE J)	FLOODING
AUXILIARY BUILDING UHI EQUIPMENT ROOMS A14, A15 EL 729.0	1	AVG 80 MAX 92 MIN 70	50 60 40	ATM ATM ATM	NA 5×10^2 NA	NA NA NA	B	NOTE L
	2	MAX 110 MIN 40 } NOTE D	90 10	ATM ATM	NA NA	$< 1 \times 10^4$ NA		
	3	NA	NA	NODE 1	NA	NA		



NOTES:

A. OPERATIONAL CONDITION DEFINITIONS

1. NORMAL
2. ABNORMAL (THIS ALSO INCLUDES ACCIDENT RADIATION LEVELS FOR LOCA CONDITIONS.)
3. TORNADO

B. ATM INDICATES A PRESSURE EQUAL TO ATMOSPHERIC PRESSURE WILL BE PRESENT. NORMAL ATMOSPHERIC PRESSURE AT WBN IS APPROXIMATELY 14.4 PSIA. ATM(+) INDICATES A PRESSURE SLIGHTLY ABOVE NORMAL, WHILE ATM(-) INDICATES A PRESSURE SLIGHTLY BELOW.

C. ALL 40 YEAR INTEGRATED RADIATION DOSES SHOWN ARE UPPER LIMITS FOR THE SUMMATION OF THE GAMMA AND BETA CONTRIBUTIONS UNLESS OTHERWISE INDICATED. RADIATION DOSE INFORMATION WAS TAKEN FROM THE FSAR SECTION 3.11.2, FIGURES 12.1-1 THROUGH 12.1-20 AND FIGURES 15.5-16 THROUGH 15.5-20. THE INTEGRATED ACCIDENT DOSES GIVEN ARE FOR A LOCA CONDITION. TOTAL RADIATION DOSE IS OBTAINED BY ADDING THE 40 YR INTEGRATED AND INTEGRATED ACCIDENT DOSES. USE $1X10^4$ RADS TOTAL 40 YR + ACCIDENT DOSE FOR QUALIFICATION OF EQUIPMENT IN AREAS WHICH HAVE $5X10^2$ RADS 40 YR INTEGRATED DOSE AND $<1X10^4$ RADS INTEGRATED ACCIDENT DOSE.

D. THESE ABNORMAL TEMPERATURES COULD OCCUR AS A RESULT OF OUTSIDE TEMPERATURE EXCURSIONS, TEMPORARY, GREATER THAN DESIGN HEAT LOAD OR DEGRADED ENVIRONMENTAL CONTROL SYSTEM OPERATION. THIS CONDITION COULD EXIST FOR UP TO 8 HOURS PER EXCURSION AND WILL OCCUR LESS THAN 1% OF THE PLANT LIFE. THESE TEMPERATURES INCLUDE THE EFFECTS OF A LOCA. (I.E. GREATER THAN NORMAL HEAT LOADS).

E. NA INDICATES NOT APPLICABLE FOR THIS OPERATIONAL CONDITION.

F. THESE SPACES WILL REACH 120°F ONCE EACH MONTH FOR 1 HOUR PLUS ONCE EVERY 18 MONTHS FOR 24 HOURS (CONTINUOUS) DURING PERIODIC TESTING OF THE DIESELS.

G. THE MAXIMUM AND MINIMUM ABNORMAL RELATIVE HUMIDITIES GIVEN WILL OCCUR LESS THAN 1% OF THE TIME WITH A DURATION OF 8 HRS FOR EACH EXCURSION. THE MAXIMUM TEMPERATURE ASSOCIATED WITH 90% RH IS 80°F. FOR TEMPERATURES ABOVE 80°F, THE SPECIFIC HUMIDITY SHALL BE ASSUMED TO BE CONSTANT AT 140 GRAINS OF MOISTURE PER POUND OF DRY AIR.

H. THE PROBABILITY OF A TORNADO IS $3.36X10^{-5}$ OR ABOUT ONCE EVERY 30,000 YRS (REF FSAR 2.3.1-3). THE FIGURE(S) LISTED DEFINE THE PRESSURE VS TIME RELATIONSHIP DURING A DESIGN BASIS TORNADO FOR THE SUBJECT ROOM(S).

I. THE MAXIMUM POSSIBLE FLOOD LEVEL OF 740.3 FT WILL NOT AFFECT THE SAFETY RELATED EQUIPMENT IN THESE AREAS. (REF. FSAR SECTION 2.4.2.2).

J. AREAS LISTED ARE DIVIDED INTO THREE CATEGORIES DEFINED AS FOLLOWS:

A. SPACES THAT ARE SERVED BY SAFETY-RELATED REDUNDANT ENVIRONMENTAL CONTROL SYSTEMS BACKED BY ONSITE EMERGENCY ELECTRICAL POWER.

B. SPACES NOT MAINTAINED BY REDUNDANT ENVIRONMENTAL CONTROL SYSTEMS BACKED BY ONSITE EMERGENCY ELECTRICAL POWER.

C. SPACES SERVED BY NON SAFETY-RELATED ENVIRONMENTAL CONTROL SYSTEMS DURING NORMAL CONDITIONS AND REDUNDANT SAFETY-RELATED SYSTEMS DURING ACCIDENT CONDITIONS.

K. THIS ABNORMAL MINIMUM TEMPERATURE WILL OCCUR FOR A DURATION OF UP TO 1 HOUR AT WHICH TIME THE SPACE TEMPERATURE WILL INCREASE TO 20°F FOR THE NEXT 8 HOURS. TEMPERATURES WILL RETURN TO THE NORMAL MINIMUM OF 40°F THEREAFTER. THESE CONDITIONS WILL EXIST LESS THAN 1% OF THE PLANT LIFE.

L. THE COMPUTED MAXIMUM FLOOD LEVEL OF 740.3 FT WILL CAUSE FLOODING TO ELEVATION 738.8 FT IN THE TURBINE, CONTROL AND AUXILIARY BUILDINGS. ALL EQUIPMENT REQUIRED [illegible] MAINTAIN PLANT SAFETY DURING THE FLOOD, AND FOR 100 DAYS AFTER THE BEGINNING OF THE FLOOD SHOULD BE EITHER DESIGNED TO OPERATE SUBMERGED, LOCATED ABOVE THE MAXIMUM FLOOD LEVEL, OR OTHERWISE PROTECTED. WIND WAVE RUNUP DURING THE PROBABLE MAXIMUM FLOOD AT THE DIESEL GENERATOR BUILDING REACHES ELEVATION 740.1 FT WHICH IS 1.9 FT BELOW THE OPERATING FLOOR ELEVATION. CONSEQUENTLY, WIND WAVE RUNUP WILL NOT IMPARE THE SAFETY FUNCTIONS OF THE DIESEL GENERATOR BUILDING. THE ELECTRICAL EQUIPMENT ROOM OF THE INTAKE PUMPING STRUCTURE WILL FLOOD AT ELEVATION 728 FT, HOWEVER, THE DESIGN BASIS WATER LEVEL FOR THE REMAINING STRUCTURE IS ELEVATION 740 FT. THE AUXILIARY AND CONTROL BUILDINGS WILL FLOOD WITH THE WATER LEVEL AT ELEVATION 729 FT.

M. THESE SPACES ARE OPEN TO THE OUTSIDE ATMOSPHERE AND AS A RESULT ARE SUBJECT TO EXTREME ENVIRONMENTAL CONDITIONS 100% RELATIVE HUMIDITY WILL EXIST UP TO 5% OF THE YEAR.

N. THIS AREA HAS NO ENVIRONMENTAL CONTROL

REFERENCE:

1. SUMMARY OF MILD ENVIRONMENTAL CONDITIONS FOR SEQUOYAH AND WATTS BAR NUCLEAR PLANTS (NEB-81-1119-235).

REV NO	ECN NO	DATE	DSGN	DRWN	CHKD	SUPV	ENGR	INSP	SUBM	RECM	APPD

DSGN / DRWN / CHKD / SUPV / INSP / ENGINEER

SCALE: EXCEPT AS NOTED

AUXILIARY BUILDING
UNITS 1 & 2

ENVIRONMENTAL DATA
ENVIRONMENT - MILD
EL 729.0

WATTS BAR NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY
DIVISION OF ENGINEERING DESIGN

SUBMITTED | RECOMMENDED | APPROVED

KNOXVILLE 8-26-83 | 85 | M | 47E235-10 | R0

RECORD DRAWING AS CONSTRUCTED

FURTHER REVIEWED
EEB
CEB
MEB
NEB

ECN-3606

INSPECTED AND APPROVED FOR ISSUE

PRINT H
SIZE F
BR OR PROJ ME EE CE AD CD ED MD SN SL PA HPP YC
PRINTS REQD-R

WBN036

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AUG 30 1983
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