TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

5N 157B Lookout Place

December 13, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. B. Youngblood, Project Director
PWR Project Directorate No. 4
Division of Pressurized Water Reactor (PWR) Licensing A
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Youngblood:

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

NRC forwarded to us the final draft technical specifications for Watts Bar Nuclear Plant unit 1 by letter dated December 11, 1984. By letters dated February 15, April 15, May 20, and September 11, 1985, NRC forwarded various revised pages to the December 11 final draft technical specifications. NRC has requested that we certify the unit 1 final draft technical specifications, as revised, accurately reflect the as-built facility, the FSAR (as amended), and the Safety Evaluation Report (SER) analysis.

We have previously provided unit 1 technical specification certifications (with exceptions) by letters dated April 9, June 21, and July 15, 1985. We have completed our review of the technical specification revisions transmitted by NRC's September 11, 1985 letter. According to the best of our knowledge, we now certify that the unit 1 technical specifications transmitted by letter dated December 11, 1984 and revised by letters dated February 15, April 15, May 20, and September 11, 1985 accurately reflect the as-built facility as described in the FSAR (as amended) and the SER except where indicated in the enclosures.

B001
1/1

ADD:
AD - J. Knight (ltr only)
EB (BALLARD)
EICSB (ROSA)
PSB (GAMMILL)
RSB (BERLINGER)
FOB (BENAROYA)

Director of Nuclear Reactor Regulation December 13, 1985

Enclosure 1 describes technical specification changes which have been previously sent to the NRC and are required for certification of the final draft technical specifications. Enclosure 2 contains new changes which resulted from the review of the technical specification revisions provided by your September 11, 1985 letter and the ongoing review of plant Surveillance Instructions.

Please note that the addition of the waste gas system to the Reactor Coolant Sources Outside Containment program (technical specification 6.8.5.a) has not been addressed in either of the enclosures. We believe that this issue will be resolved before issuance of a full-power license and the waste gas system will be removed from the program. Therefore, no plant instructions will be prepared to address this requirement.

If there are any questions, please get in touch with K. P. Parr at FTS 858-2680.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. W. Hufham
Manager of Licensing

Sworn to and subscribed before me
this 13th day of Dec. 1985

Paulette H. White
Notary Public

My Commission Expires 8-24-88

Enclosures

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

ENCLOSURE 1

Technical Specifications
Previously Submitted

Natural Circulation Testing

By letter dated May 30, 1984 (attached), TVA requested that certain Technical Specifications 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.

Please note that this request was included in the letter to NRC dated April 9, 1985 (certification).

.127 840530 005 ℓ10

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) Docket Nos. 50-390
Tennessee Valley Authority)

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.

~~8406040270~~ 2pp.

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, 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).

Low Temperature Overpressure Protection (LTOP)

Specification 3.1.2.3 for boron injection flow paths should be revised to delete the surveillance requirement that requires charging pumps to be inoperable. The reasons for this change are threefold. First, the basis for this specification is not consistent with the surveillance requirement on pump inoperability. The specification is designated for ensuring the operability of boration capability during shutdown. The surveillance requirements on the other hand, is designed to ensure that the assumptions in he LTOP analysis are met. Second, the action statement does not address appropriate remedial actions for when the requirements of the surveillance are not met. Third, the applicability of the specification includes mode 6. LTOP events are not credible when the reactor vessel head is removed in mode 6. However, the surveillance requirements is applicable in mode 6 regardless of whether the reactor vessel head is on or off.

Specification 3.4.9.3 for LTOP should be revised to include an action statement to define remedial actions for the cases where one or more pumps are not inoperable or isolated from the reactor coolant system. The surveillance requirement on verifying pump inoperability or isolation from the reactor coolant system should be added. These changes properly put all activity associated with LTOP in one technical specification.

Specification 3.5.3 should be revised to delete the footnote and surveillance requirement associated with LTOP. The footnote is inconsistent with technical specification 3.1.2.4 and 3.5.2. These two specifications require that two charging pumps be operable in mode 3. Mode 3 is defined as an average reactor coolant temperature greater than or equal to 350^oF. The cold leg temperatures can and will be below 350^oF when the plant is in mode 3. Specification 3.5.3, however, requires that all but one charging pump be inoperable whenever the cold leg temperatures are less than or equal to 350^oF. The situation can exist where the pumps are simultaneously required to be inoperable and operable. The requirement will be violated each time a transition between modes 3 and 4 is made; therefore, this problem must be resolved for certification of the technical specifications. Furthermore, this footnote is applicable only in mode 4 and the footnote is the only place where controls are put on the safety injection pumps.

The bases for specifications 3/4.1.2 and 3/4.5.3 should be revised to delete all references to LTOP. The bases for LTOP should be revised to accurately reflect the applicability of specification 3.4.9.3. With this change, the bases for specification 3/4.4.9 adequately address the bases for LTOP.

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LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that a discharge pressure of greater than or equal to 2400 psig is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are 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.

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a nominal lift setting less than or equal to that shown in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 3 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 310°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 3 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 3 square inch vent within 8 hours.

d ~~c~~. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e ~~d~~. The provisions of Specification 3.0.4 are not applicable.

c. With one or more of the pumps required to be inoperable or isolated not inoperable or isolated from the RCS, make the pump(s) inoperable or isolated from the RCS within 12 hours or depressurize and vent the RCS through at least a 3 square inch vent within the following 8 hours

FINAL DRAFT

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

4.4.9.3.3 All charging pumps, excluding those required by specifications 3.1.2.3 and 3.5.3, and safety injection pumps shall be demonstrated inoperable at least once per 12 hours by verifying that 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.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

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

A maximum of one centrifugal charging pump 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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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 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.

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

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

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

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

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

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 3 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle reactor coolant pump with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures (heat input), or (2) the start of a HPSI pump and its injection into a water solid RCS (mass input).

The Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance of the LTOPS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one Safety Injection pump and all but one centrifugal charging pump while in ~~MODES 4, 5, and 6~~ with the reactor vessel head installed and disallow start of a RCP if secondary temperature is more than 50°F above primary temperature.

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

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the

MODE 4 when one or more of the RCS cold legs are less than or equal to 310°F and MODES 5 and 6

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ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump and one Safety Injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging 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.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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

Tables 3.3-6 and 4.3-3

By letter dated February 16, 1985 TVA requested an exemption to the requirements of 10 CFR Part 70.24. As stated in the letter, the exemption request was made to support proposed changes to the draft technical specifications that deleted the requirements for a criticality monitor. This exemption request was submitted because NRC informed TVA that the current exemption to 10 CFR Part 70.24 contained in the Special Nuclear Materials (SNM) license would expire upon issuance of the operating license.

By letter dated April 9, 1985 (cerification) TVA resubmitted the proposed technical specification changes that deleted the requirements for a criticality monitor and referenced the exemption request as justification. The technical specification changes included in this letter were incomplete as a change to the applicability was omitted.

A correct set of technical specification changes is attached. The changes are identical to those previously provided by letter dated January 30, 1985. The changes consists of two parts: deletion of criticality from the monitor functional description and limiting the applicability of the specificaton to only when irradiated fuel is in the spent fuel pit.

By letter dated April 15, 1985, NRC approved the exemption to 10 CFR Part 70.24. However, the NRC limited the period of exemption to when irradiated fuel is placed in the spent fuel pit. NRC has indicated that extensions to the exemption period would not be granted without further analytical justification. We believe it is necessary to restate the basis for the technical specification changes and provide additional arguments to support the extension of the exemption to 10 CFR Part 70.24 for the life of the plant.

Radiation monitors RE-90-102 and RE-90-103 are designed to perform two functions: detection and isolation initiation for a fuel handling accident in the Auxiliary Building and criticality monitor. The first protection function is required whenever irradiated fuel is in the spent fuel pit. This protection function is not required when only new fuel is stored in the spent fuel pit because there is no potential for radioactive release.

The second protection function is required by 10 CFR Part 70.24. The limiting point in time for criticality concerns exists when only new fuel is stored in the spent fuel pit. This limiting case has been analyzed and found to present no criticality problems. As stated in FSAR section 9.1.2.3, the design of the spent fuel racks are "sufficient to ensure a Keff less than or equal to 0.95 even if unborated water is used to fill the spent fuel storage pool." The details of the calculations are described in FSAR section 4.3.2.7. The FSAR states, "in the analysis for the storage facilities, the fuel assemblies are assumed to be in their most reactive condition, namely fresh or undepleted and with no control rods or removable neutron absorbers present." The important aspects of the analytical assumptions are incorporated into the technical specifications to preserve the validity of the calculations. The maximum enrichment and fuel design specifications are listed in technical specification 5.3.1. The design specifications for the spent fuel racks are listed in technical specifications 5.6.1.1 and 5.6.3.

NRC approved an exemption to 10 CFR 70.24 for the period of time up to the storage of spent fuel in the spent fuel pit by the previously mentioned letter of April 15, 1985. Only new fuel will be stored in the spent fuel pit during this period. The FSAR documents that the storage of new fuel is the most limiting condition for criticality concerns. In effect, the NRC has approved the exemption to 10 CFR Part 70.24 for the most limiting case already. An extension to the exemption for the life of the plant should require no further analysis or justification.

It should be noted that any change to the design of the fuel, enrichment of the fuel, or design of the storage racks would require an evaluation under the requirements of 10 CFR Part 50.59. Any change to these features that alters the validity of the current FSAR analysis would require a technical specification change.

The technical specifications should be revised to delete any requirements for a criticality monitor based on the limiting case FSAR analysis and the exemption to 10 CFR Part 70.24. RE-90-102 and RE-90-103 should only be required to be operable when irradiated fuel is stored in the spent fuel pit.

The net effect in the operation of Watts Bar, with the exemption to 10 CFR Part 70.24 for the life of the plant and the corresponding technical specification changes, would be to eliminate the requirements for RE-90-102 and RE-90-103 for the first cycle of operation for unit 1. RE-90-103 would be required to be operable once the spent fuel is transferred to the spent fuel pit during the first refueling outage. The benefits to TVA include the manpower and cost savings associated with not requiring maintenance and surveillance on these monitors during the first cycle of operation. There is a corresponding savings in paperwork costs because any work associated with these monitors is controlled with specific recordkeeping requirements. In addition, receipt of unit 1 cycle 2 fuel and fuel loading operations associated with unit 1 could proceed without application of the monitoring requirements and associated action statements imposed by the technical specifications.. An inoperable monitor would not disrupt or delay these activities.

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)		2	*	≤ 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 3.3-6 (Continued)

TABLE NOTATIONS

*With fuel in the irradiated fuel storage areas.

**400 cpm is equivalent to 1 x 10^{-5} μCi/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.

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FEB 15 1985

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				
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 irradiated fuel in the fuel storage areas.

Technical Specification Table 4.3-6

Surveillance requirement 4.3.3.5 of the Watts Bar unit 1 technical specifications requires that each remote shutdown instrument be demonstrated operable by performing a channel check and channel calibration at the frequencies shown in Table 4.3-6. For the source range flux detectors, the channel check is required to be performed monthly; however, above the P-6 interlock, the high-voltage power supply to the source range detectors is removed. Therefore, a channel check above P-6 will not demonstrate operability of the source range flux detectors. TVA requests that a footnote be added to Table 4.3-6, item 1, to not require the source range channel check above the P-6 interlock. This same footnote already exists on the source range detector surveillance requirements in Table 4.3-1, item 6. Attached is a marked-up copy of Table 4.3-6 showing the necessary change.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Source Range Nuclear Flux	M##	R*
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Temperature - Hot Leg	M	R
4. Pressurizer Pressure (Wide Range)	M	R
5. Pressurizer Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Level	M	R
8. Control Rod Bottom Bistables	M	R
9. RHR Flow Rate	M	R
10. RHR Temperature	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Pressurizer Relief Tank Pressure	M	R
13. Containment Pressure	M	R

*Neutron detectors may be excluded from CHANNEL CALIBRATION.
Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint

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Surveillance Requirement 4.4.3.3

Surveillance requirement 4.4.3.3 requires that the emergency power supply for the pressurizer heaters be demonstrated OPERABLE once per 18 months by manually transferring power from the normal to the emergency power supply. At Watts Bar Nuclear Plant the normal and emergency power supply for the heaters are one in the same: the 6.9 kV shutdown boards. It is the power supply for the 6.9 kV shutdown boards that changes; offsite power is the preferred source and diesel generators are the emergency source. As part of the 18-month diesel generator testing (SR 4.8.1.1.2.f.4) a blackout signal is simulated. Load shedding and subsequent re-energization of emergency loads are verified. At this time the emergency loads are powered from the emergency power source.

This test satisfies the intent of surveillance requirement 4.4.3.3 by demonstrating the operability of the emergency power supply to the pressurizer heaters. Therefore, TVA requests the word "manually" be deleted from surveillance requirement 4.4.3.3.

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3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

Surveillance Requirement 4.4.4.3.a

Surveillance requirement 4.4.4.3.a requires that the PORVs and block valves be demonstrated OPERABLE once per 18 months by manually transferring power from the normal to the emergency power supply. At Watts Bar Nuclear Plant the normal and emergency power supply for the PORVs and block valves are one in the same: the 6.9 kV shutdown boards. It is the power supply for the 6.9 kV shutdown boards that changes; offsite power is the preferred source and diesel generators are the emergency source. As part of the 18-month diesel generator testing (SR 4.8.1.1.2.f.4) a blackout signal is simulated. Load shedding and subsequent re-energization of emergency loads are verified. At this time the emergency loads are powered from the emergency power source.

This test satisfies the intent of surveillance requirement 4.4.4.3.a by demonstrating the operability of the emergency power supply to the PORVs and block valves. Therefore, TVA requests the word "manually" be deleted from surveillance requirement 4.4.4.3.a.

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. of Specification 3.4.4.

4.4.4.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. ~~Manually~~ transferring motive and control power from the normal to the emergency power supply, and

b. Operating the valves through a complete cycle of full travel.

Technical Specification 4.5.2.h.1 - Charging Pump Flow

By letter dated June 21, 1985, TVA provided information to NRC on the revised ECCS analysis for Watts Bar. The revised analysis included the CCP miniflow lines being open on an SI signal. The technical specification changes associated with the revised analysis must be resolved for TVA to certify the Watts Bar unit 1 technical specifications.

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

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

   a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to ~~346~~ 315 gpm, and

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

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

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

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

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

Surveillance Requirement 4.6.1.8.1.d.4

By letter dated June 21, 1984, TVA provided information to NRC on the analysis performed to support an annulus in leakage rate of 500cfm. Also included in the June 21, 1984 letter was a technical specification change associated with the analysis.

SURVEILLANCE REQUIREMENTS (Continued)

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

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.2%,

d. At least once per 18 months, by:

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

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

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

4) Verifying that the air cleanup subsystem maintains the annulus building at a pressure equal to or more negative than minus 0.5 inches Water Gauge relative to the Mechanical Equipment Room with an inleakage of less than or equal to ~~100~~ 500 cfm, and

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

e. After each complete or partial replacement of a HEPA filter bank by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

4.6.1.8.2 The annulus pressure shall be verified to be equal to or more negative than minus 5.0 inches Water Gauge with respect to the Mechanical Equipment Room at least once per 24 hours.

SURVEILLANCE REQUIREMENT 4.8.1.1.2.f.1

TVA is proposing that the wording for the surveillance requirement be revised to more accurately reflect the diesel generator inspection program at Watts Bar. The inspections currently recommended by the manufacturer occur at a variety of intervals from monthly to every five years. No specific inspections are specified at an 18-month interval.

TVA plans to verify that the diesel has been subjected to inspections in accordance with the manufacturer's recommendation of an 18-month interval to satisfy the surveillance requirement. This verification will be done by a review of the maintenance logs.

The surveillance requirement should be reworded to accurately reflect TVA's plans. This problem was identified by the NRC-OIE Region II inspection team. The TVA approach seemed accepteble to them; however, they indicated that the literal interpretation of the current wording could lead to interpretation problems.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification;

c) A flash point equal to or greater than 125°F; and

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

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

e. At least once every 31 days by obtaining a sample of fuel oil from each of the four interconnected tanks which constitute the 7-day fuel oil storage tanks in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;***.

f. At least once per 18 months by:

1) ~~Subjecting the diesel to an~~ Verifying that the diesel has been subjected to inspections in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

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

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

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

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

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

***For the first 6-month period following issuance of the low power license, only two of the four interconnected tanks are required to be sampled every 31 days for each 7-day fuel oil storage tank.

Technical Specifications
3.8.3.1 and 3.8.3.2

Recently the NRC has questioned components in the Watts Bar electrical distribution system where redundant power divisions can be interconnected. At Watts Bar the 120-volt a.c. vital inverters can be powered from either an A or B train 480-volt shutdown board and receive emergency power from the 125-volt d.c. vital battery board associated with that inverter. Currently the Watts Bar technical specifications require the inverter be connected to the emergency power supply, but there are no requirements to be connected to the associated division 480-volt shutdown board. Therefore, redundant power divisions could be interconnected at the inverter. Attached are proposed changes to technical specifications 3.8.3.1 and 3.8.3.2 which will require the inverter to be powered from its associated 480-volt power division.

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3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The Shutdown Board Room chillers shall be OPERABLE and the following electrical busses shall be energized in the specified manner with tie breakers open both between redundant busses within the unit and between units at the same station:

a. Train A-A.C. Emergency Busses consisting of:

6900-Volt Shutdown Board 1A-A,
480-Volt Shutdown Board 1A1-A,
480-Volt Shutdown Board 1A2-A,
6900-Volt Shutdown Board 2A-A,
480-Volt Shutdown Board 2A1-A, and
480-Volt Shutdown Board 2A2-A.

b. Train B-A.C. Emergency Busses consisting of:

6900-Volt Shutdown Board 1B-B,
480-Volt Shutdown Board 1B1-B,
480-Volt Shutdown Board 1B2-B,
6900-Volt Shutdown Board 2B-B,
480-Volt Shutdown Board 2B1-B, and
480-Volt Shutdown Board 2B2-B.

c. 120-Volt A.C. Vital Channels 1-I and 2-I energized from its associated inverter, connected to D.C. Channel I and to the associated 480-volt power division;*

d. 120-Volt A.C. Vital Channels 1-II and 2-II energized from its associated inverter, connected to D.C. Channel II and to the associated 480-volt power division;*

e. 120-Volt A.C. Vital Channels 1-III and 2-III energized from its associated inverter, connected to D.C. Channel III and to the associated 480-volt power division;*

f. 120-Volt A.C. Vital Channels 1-IV and 2-IV energized from its associated inverter, connected to D.C. Channel IV and to the associated 480-volt power division;*

g. 125-Volt D.C. Board I energized from Vital Battery Bank I;

h. 125-Volt D.C. Board II energized from Vital Battery Bank II;

i. 125-Volt D.C. Board III energized from Vital Battery Bank III; and

j. 125-Volt D.C. Board IV energized from Vital Battery Bank IV.

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

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*Two inverters may be disconnected from their D.C. Bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) the vital busses are energized; and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

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LIMITING CONDITION FOR OPERATION

ACTION:

a. With one of the required trains of A.C. emergency busses not fully energized, reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus and its associat[ed] 480-v[olt] power divis[ion] within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

and its associated 480-volt power division

c. With one D.C. Bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one train of the Shutdown Board Room chillers inoperable, restore the inoperable chillers to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

4.8.3.1.2 The Shutdown Board Room chillers shall be determined OPERABLE at least once per 31 days by verifying that each train is started and operates for at least 15 minutes.

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ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. emergency busses consisting of two 6900-volt and four 480-volt A.C. emergency busses;

b. Two 120-volt A.C. vital busses, either Channels 1-I and 1-III, or 1-IIand 1-IV energized from their associated inverters connected to their respective D.C. channels and to their associated 480-volt power division; and

c. Two 125-volt D.C. busses, either Channels I and III, or II and IV energized from its associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours depressurize and vent the RCS through at least a 3 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

Technical Specification 3.8.4.2

In a February 14, 1985, letter from NRC, changes to technical specification 3.8.4.2 were approved for the final draft of the Watts Bar unit 1 technical specifications. These changes were made to provide separate requirements for motor-operated valves with thermal overload devices which are not bypassed for all unit 1 accident conditions. Since these changes were approved, the thermal overload devices for the valves in Table 3.8-3 have been modified so that the thermal overloads are bypassed for all unit 1 accident conditions. Therefore, the valves in Table 3.8-3 should be included in Table 3.8-2, and Table 3.8-3 should be deleted. Also, the requirements in technical specification 3.8.4.2 associated with Table 3.8-3 should be deleted. Attached are marked-up technical specification pages showing the necessary changes.

0298K

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FEB 15 1995

MOTOR-OPERATED VALVES THERMAL OVERLOAD BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload bypass devices, integral with the motor starter of each valve listed in Table 3.8-2 ~~and Table 3.8-3~~ shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

a. With the thermal overload protection for one or more of the Table 3.8-2 valves not bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) of the affected system(s).

~~b. With the thermal overload protection for one or more of the Table 3.8-3 valves inoperable, bypass the inoperable thermal overload within 8 hours; restore the inoperable thermal overload to OPERABLE status within 30 days or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).~~

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload bypass devices shall be demonstrated OPERABLE:

~~a.~~ At least once per 92 days and following maintenance on the motor starter for the motor operated valves in Table 3.8-2 by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST of the bypass circuitry. ~~and~~

~~b. At least once per 18 months and following maintenance on the motor starter for the motor operated valves in Table 3.8-3 by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overload devices for the valves in Table 3.8-3.~~

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD 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
2-FCV-67-66	DG Ht Ex	
2-FCV-67-67	DG Ht Ex	Yes
0-FCV-67-152	CCWS Ht Ex Throttling	Yes
2-FCV-67-65	DG Ht Ex	
2-FCV-67-68	DG Ht Ex	

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TABLE 3.8-3

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

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

Delete

Technical Specification 3.8.4.3

During meetings held April 2, 10, and 12, 1985, TVA and NRC representatives discussed those components at Watts Bar assumed to be either (1) deenergized directly or indirectly by an Engineered Safety Feature Actuation System (ESFAS) signal, or (2) administratively deenergized during normal operation. These components were not included in the submerged components analysis; therefore, the NRC staff believed a formal technical specification was needed.

By letter dated April 17, 1985, TVA proposed a technical specification to cover those components deenergized from an ESFAS signal (category 1 above). The administratively deenergized components (category 2 above) had not been included in the original analysis. However, the analysis performed following the April meeting did not take credit for these administrative controls. The analysis was performed considering these circuits energized.

By letter dated May 20, 1985, NRC transmitted to TVA the new technical specification 3.8.4.3. Included in a specification 3.8.4.3 is a requirement to verify that valves 1-FCV-74-1 and 1-FCV-74-9 are deenergized. These valves are administratively deenergized components that were included in the subsequent analysis. No technical specification requirements are necessary for these valves. Attached is a marked-up page showing the changes necessary to make specification 3.8.4.3 agree with the April 17, 1985, TVA proposal and current analyses. This change must be made for Watts Bar to certify that the technical specifications agree with the plant design and submerged component analysis.

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SUBMERGED COMPONENT CIRCUIT PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.3 The submerged component circuits associated with valves 1-FCV-74-1 and 1-FCV-74-9, and with each component listed in Table 3.8-4 shall be OPERABLE.

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

ACTION:

With one or more submerged components circuits inoperable, restore the inoperable circuit to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The submerged component circuits shall be demonstrated OPERABLE:

~~a. At least once per 31 days by verifying that valves 1-FCV-74-1 and 1-FCV-74-9 are de-energized, and~~

~~b.~~ At least once per 18 months by verifying that the components listed in Table 3.8-4 are automatically de-energized on a simulated accident signal.

ENCLOSURE 2

New Certification Changes

0038B

Technical Specification Section 3/4.3

The Nuclear Regulatory Commission approved changes dated August 30, 1985, deleting specification 3.3.3.7 (chlorine detection systems) from the final draft of the Watts Bar unit 1 technical specifications. Also, all of the specifications from 3.3.3.8 through 3.3.3.11 were renumbered. This renumbering has made several references in other sections of the technical specifications incorrect. Also, Watts Bar is using a numbering system for Surveillance Instructions (SI) which correlates surveillance requirement numbers to SI numbers. The renumbering of Section 3/4.3 will require numerous SIs to be modified. The best way to correct these problems would be to substitute the last specification in section 3/4.3 (turbine overspeed protection, 3.3.4) for the deleted chlorine detection system specification and return the specification numbers for 3.3.3.8 through 3.3.3.11 back to the previous numbers. This change must be made for Watts Bar to certify the final draft technical specifications. If issued in the current order the technical specifications will not agree with the surveillance instructions.

Surveillance Requirement 4.3.2.2

The ENGINEERED SAFETY FEATURES RESPONSE TIME testing for the turbine driven auxiliary feedwater pump must be performed in mode 3 or above for the steam supply to be sufficient for the pump to deliver rated discharge pressure. Therefore, an exclusion should be added to surveillance requirement 4.3.2.2 to allow entry into mode 3 without the response time test being performed. Attached is a marked-up copy of page 3/4 3-15 of the Watts Bar unit 1 technical specifications showing the proposed change.

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3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

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

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. The provisions of specification 4.0.4 are not applicable for entry into mode 3 for the response time testing of the turbine driven auxilliary feedwater pump.

Technical Specification Table 3.3-5

The footnotes applied to items 9.a and 12 of technical specification Table 3.3-5 indicate that the response time test for the auxiliary feedwater pumps, when started by a steam generator water level-high-high signal or the trip of all main feedwater pumps, includes a diesel generator start. This does not agree with the starting logic for the auxiliary feedwater pumps. A steam generator water level-high-high signal or a trip of all main feedwater pumps will start the auxiliary feedwater pumps only if a black-out or a safety injection signal is not present, therefore, the start of the pumps is not dependent on the start of the diesel generators for these two signals. To accurately describe the Watts Bar system, note 1 on item 9.a and note 10 on item 12 should be deleted. Attached is a marked-up copy of Table 3.3-5 showing the necessary changes.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME ~~IN~~ SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12[5]/22[4]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8[3]
3) Containment Isolation-Phase "A"[6]	≤ 18[2]/28[1]
4) Containment Ventilation Isolation	N.A.
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
b. Steam Line Isolation	≤ 7
7. Containment Pressure--High-High	
a. Containment Spray	≤ 147[1]
b. Containment Isolation-Phase "B"	≤ 71[2]/81[1]
c. Steam Line Isolation	≤ 7
d. Containment Air Return Fans	≤ 660
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 11[3]
9. Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60[7] ~~(1)~~
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 60[8]
10. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
Automatic Switchover to Containment Sump	≤ 250
11. Loss-of-Offsite Power	
Auxiliary Feedwater Pumps	≤ 60[10]
12. Trip of All Main Feedwater Pumps	
Auxiliary Feedwater Pumps	≤ 60 ~~(10)~~

Technical Specification Table 3.3-11

The description of the fire detectors for zones 332 and 333 in Table 3.3-11 should be revised. Currently the detectors are listed as smoke detectors. The detectors in zones 332 and 333 are heat detectors. Attached is a marked-up copy of technical specification page 3/4 3-73 showing the necessary changes.

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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
330	Pipe Chase, U-1, El. 737, 713, 692			20/0
332	North Main Stm. Vlv. Rm., El. 737	9/0		~~9/0~~
333	South Main Stm. Vlv. Rm., El. 737	10/0		~~10/0~~
455	Post Accident Samp. Fac., U-1, El. 737			0/2
456	Post Accident Samp. Fac., U-1, El. 737			0/2
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

Technical Specification Table 3.3-13

Specification 3.3.3.9 of the Watts Bar unit 1 technical specifications requires that the gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 be OPERABLE with their alarm/trip setpoints set to ensure the limits of specification 3.11.2.5 are not exceeded. This requirement is applied to the hydrogen and oxygen monitors of the waste gas holdup system and is intended to indicate a potential explosive gas mixture. At Watts Bar the alarm for the explosive gas monitoring system is only on oxygen concentration and is set for 2 percent oxygen by volume. TVA believes that this alarm meets the intent of specifications 3.3.3.9 and 3.11.2.5; however, to prevent future enforcement problems, a note should be added to item 2 of Table 3.3-13. Attached is a marked-up copy of Table 3.3-13 showing the proposed changes.

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 (Low Range)	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 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

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

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

The alarm for explosive gas mixture is only on oxygen concentration

ACTION STATEMENTS

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

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

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

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

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

Technical Specification Table 4.3-9

Technical specification Table 4.3-9 item 5.a should be revised to make the notes adequately address the design of the radiation monitor. The noble gas activity monitor for the auxiliary building ventilation and the fuel handling area ventilation systems provides a control room annunciation for any of the following three conditions:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

However, automatic isolation of this pathway only occurs if the instrument indicates measured levels above the alarm/trip setpoint. Therefore, the notes for item 5.a under "Analog Channel Operation Test" should be "(2)(6)". Attached is a marked-up copy of technical specification page 3/4 3-87 showing the necessary change.

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)~~(2)(6)	*
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(2)(6)	*

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Technical Specification Table 3.8-1

By letter dated September 11, 1985, NRC transmitted to TVA approved changes to Table 3.8-1; however, several errors were identified when the changed pages were reviewed. Attached are marked-up pages showing the changes necessary to make Table 3.8-1 accurately reflect the as-built facility.

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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 -1/D17(+)	FU-236 -1/D17(-)	125V VI-BATT BDI PNL 4	RCP 1 SEAL RTN FLOW CONT VLV
FU-236 -1/D20(+)	FU-236 -1/D20(-)	125V VI-BATT BDI PNL 4	RCS PRESS RLF TK VT
FU-236 -1/D21(+)	FU-236 -1/D21(-)	125V VI-BATT BDI PNL 4	RCS PRESS RLF TK DR
FU-236 -1/D22(+)	FU-236 -1/D22(-)	125V VI-BATT BDI PNL 4	GLYCOL SPLY FROM EXP TK
FU-236 -1/D23(+)	FU-236 -1/D23(-)	125V VI-BATT BDI PNL 4	SIS ACCUM TK 3 FILL VLV
FU-236 -1/D24(+)	FU-236 -1/D24(-)	125V VI-BATT BDI PNL 4	SIS ACCUM TK 1 FILL VLV
FU-236 -1/D27(+)	FU-236 -1/D27(-)	125V VI-BATT BDI PNL 4	RHR SPLY TEST LINE VLV
FU-236 -1/D28(+)	FU-336 -1/D28(-)	125V VI-BATT BDI PNL 4	SIS PMP OTLT TO SIS TEST LINE
FU-236 -1/D29(+)	FU-236 -1/D29(-)	125V VI-BATT BDI PNL 4	SIS CLR 3 CK VLV ISLN
FU-236 -1/D30(+)	FU-236 -1/D30(-)	125V VI-BATT BDI PNL 4	SIS ACCUM TK 1 N_2 MKUP VLV
FU-236 -1/D31(+)	FU-236 -1/D31(-)	125V VI-BATT BDI PNL 4	RCS LP 1 HT LEG FD TEST LINE VLV
FU-236 -1/D32(+)	FU-236 -1/D32(-)	125V VI-BATT BDI PNL 4	SIS FL TO COLD LEG CK VLV TEST
FU-236 -1/D33(+)	FU-236 -1/D33(-)	125V VI-BATT BDI PNL 4	RCP 1 STD PIPE MAKEUP WTR VLV
FU-236 -1/D37(+)	FU-236 -1/D37(-)	125V VI-BATT BDI PNL 4	SIS PMP OTLT TEST LINE VLV
FU-236 -1/D38(+)	FU-236 -1/D38(-)	125V VI-BATT BDI PNL 4	RHR RTN FROM SIS SMPL LINE 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/A18(+)	FU-236 -2/A18(-)	125V VI-BATT BDII PNL 4	GLYCOL RTN ISLN VLV
FU-236 -2/A20(+)	FU-236 -2/A20(-)	125V VI-BATT BDII PNL 4	PRESS GAS ISLN VLV
FU-236 -2/A21(+)	FU-236 -2/A21(-)	125V VI-BATT BDII PNL 4	PRESS LIQ ISLN VLV
FU-236 -2/A22(+)	FU-236 -2/A22(-)	125V VI-BATT BDII PNL 4	RCS HOT LEG LP 1 OR 3 ISLN VLV
FU-236 -2/A23(+)	FU-236 -2/A23(-)	125V VI-BATT BDII PNL 4	ACCUM TKS ISLN VLV
FU-236 -2/A24(+)	FU-236 -2/A24(-)	125V VI-BATT BDII PNL 4	REAC CLT DRN TK TO GAS ANAL ISLN VLV
FU-236 -2/A41(+)	FU-236 -2/A41(-)	125V VI-BATT BDII PNL 4	RCP MTR CLR B SUP VLV
FU-236 -2/A42(+)	FU-236 -2/A42(-)	125V VI-BATT BDII PNL 4	RCP MTR CLR D SUP VLV
FU-236 -2/A43(+)	FU-236 -2/A43(-)	125V VI-BATT BDII PNL 4	REAC BLDG SMP PMP DISCH ISLN VLV
FU-236 -2/A44(+)	FU-236 -2/A44(-)	125V VI-BATT BDII PNL 4	REAC CLT DR TK PMPS DISCH ISLN VLV
FU-236 -2/A45(+)	FU-236 -2/A45(-)	125V VI-BATT BDII PNL 4	REAC CLT DR TK TO VT HDR ISLN VLV
FU-236 -2/B3(+)	FU-236 -2/B3(-)	125V VI-BATT BDII PNL 4	CHGR FL RCS CL LP 1
FU-236 -2/B4(+)	FU-236 -2/B4(-)	125V VI-BATT BDII PNL 4	RES FLOW CONT VLV
FU-236 -2/B5(+)	FU-236 -2/B5(-)	125V VI-BATT BDII PNL 4	RCS PRESS PWR RLF 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/B27(+)	FU-236 -2/B27(-)	125V VI-BATT BDII PNL 4	CNTMT BLDG LWR COMPT AIR MON ISLN VLV
FU-236 -2/B28(+)	FU-236 -2/B28(-)	125V VI-BATT BDII PNL 4	CNTMT BLDG LWR COMPT AIR MON ISLN VLV
FU-236 -2/B33(+)	FU-236 -2/B33(-)	125V VI-BATT BDII PNL 4	CNTMT BLDG UPR COMPT AIR MON ISLN VLV
FU-236 -2/B36(+)	FU-236 -2/B36(-)	125V VI-BATT BDII PNL 4	CNTMT BLDG UPR COMPT AIR MON ISLN VLV
FU-236 -2/C5(+)	FU-236 -2/C5(-)	125V VI-BATT BDII PNL 4	INSTR RM COOL UNIT B VLV
FU-236 -2/C6(+)	FU-236 -2/C6(-)	125V VI-BATT BDII PNL 4	INSTR RM COOL UNIT B VLV
FU-236 -2/C7(+)	FU-236 -2/C7(-)	125V VI-BATT BDII PNL 4	CNTMT ANNS DIFF PRESS ISLN VLV
FU-236 -2/C10(+)	FU-236 -2/C10(-)	125V VI-BATT BDII PNL 4	LOCA H_2 CNTMT ISLN VLV
FU-236 -2/C11(+)	FU-236 -2/C11(-)	125V VI-BATT BDII PNL 4	LOCA H_2 CNTMT ISLN VLV
FU-236 -2/C17(+)	FU-236 -2/C17(-)	125V VI-BATT BDII PNL 4	STM GEN BLDN ISLN VLV LP 2
FU-236 -2/C18(+)	FU-236 -2/C18(-)	125V VI-BATT BDII PNL 4	UPR COMPT PURG ISLN VLV & EXH ISLN VLV
~~FU-236 -2/C21~~ (Delete)	~~52-236 -312/II~~	~~125V VI-BATT BDII PNL 4~~	~~DNSTR EXCESS LTDN HX TO HOT SMPLG RM ISLN VLV~~
FU-236 -2/C24(+)	FU-236 -2/C24(-)	125V VI-BATT BDII PNL 4	FLOOR CLG GLYCOL INLET 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
6. 120VAC BDS (Continued)			
FU-77 -L2/LF17(+)	FU-77 -L2/LF17(-)	PNL-77-L2	REAC CLNT DR TK LVL CNTL VLV
FU-275 -R76/I1, I2	52-235 -12/1-III	AUX RLY PNL I-R-76	INCORE INST A RM A/C AIR FL
FU-275 -R76/I23, I24	52-235 -5/1-IV	AUX RLY PNL I-R-76	INCORE INST B RM A/C AIR FL
FU-228 -152/10B	52-228 -152/10	JB 4806	LTG CIR IN PERS ACC LOCK EL 713
FU-228 -156/6C	52-228 -15~~2~~/6 6	JB 4810	LTR CIR IN PERS ACC LOCK EL 757
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 MTR 2 PROT
FU-202 -C2/1(+)	FU-202 -C2/1(-)	RCP BD 1C/2	RCP MTR 3 PROT
FU-202 -D3/1(+)	FU-202 -D3/1(-)	RCP BD 1D/3	RCP MTR 4 PROT
8. 48VDC BDS			
FUSE 1	FUSE 1, BAY 37, PNL L	COMM BD, BAY 39, PNL M	LOUDSPKR
FUSE 2	FUSE 1, BAY 37, PNL L	COMM BD, BAY 39, PNL M	LOUDSPKR
FUSE 3	FUSE 1, BAY 37, PNL L	COMM BD, BAY 39, PNL M	LOUDSPKR
FUSE 4	FUSE 1, BAY 37, PNL L	COMM BD, BAY 39, PNL M	LOUDSPKR
FUSE 5	FUSE 1, BAY 37, PNL L	COMM BD, BAY 39, PNL M	LOUDSPKR