TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

September 25, 1981

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

Dear Ms. Adensam:

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

Enclosed for NRC review are responses to the following NRC questions on Watts Bar Nuclear Plant.

22.60	40.118
31.44(9) - Refers to question 31.44(25)	40.122
31.44(25)	121.11
31.25	121.12
31.49	121.13
31.149	121.14
40.108	121.15
40.110	121.17
40.111	212.97
40.114	413.11
	413.12

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 25th day of Sept. 1981
Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

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Enclosure

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ENCLOSURE

WBNP-45

31.149 Question:

The analyses reported in Chapter 15 of the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents.

Based on the conservative assumptions made in defining these design-basis events and the detailed review of the analysis by the staff, it is likely that they adequately bound the consequences of single control system failures.

To provide assurance that the design basis event analyses adequately bound other more fundamental credible failures you are requested to provide the following information:

(1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

(3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors. The sensors considered should include, but should not necessarily be limited to, common hydraulic headers or impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

(4) Provide justification that any simultaneous malfunctions of the control systems identified in (2) and (3) resulting from failures or malfunctions of the applicable common power source or sensor are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

Response:

INTRODUCTION

The evaluation required to answer Question 31.149 consists of postulating failures which affect the major NSSS control systems and demonstrating that for each failure the resulting event is within the bounds of existing accident analyses. The events considered are:

a) Loss of any single instrument.
b) Break of any common instrument line.
c) Loss of power to any systems or equipment arrangements which are related by receiving power from a common single power source. Specifically, the failures are the loss of power to any inverter, to any protection set, to any control group, or to any process rack.

The analysis is conducted for all six major NSSS control systems:

1) Reactor control system.
2) Steam dump system.
3) Pressurizer pressure control system.
4) Pressurizer level control system.
5) Feedwater control system.
6) Feedwater pump speed control system.

The initial conditions for the analysis are assumed to be anywhere within the full operating power range of the plant (i.e. 0-100%) where applicable .

The results of the analysis indicate that, for any of the postulated events considered in a) through c) above, the Condition II accident analyses given in Chapter 15 of the Watts Bar FSAR are bounding.

LOSS OF ANY SINGLE INSTRUMENT

Table 1, Loss Of Any Single Instrument, is a sensor-by-sensor evaluation of the effect on the control systems itemized above caused by a sensor failing either high or low. The particular sensor considered is given, along with the number of channels which exist, the failed channel, the control systems impacted by the sensor, the effects on the control systems for failures in both directions, and the bounding FSAR accident. Where no control action occurs or where control action is in a safe direction, no bounding accident is given.

The table clearly shows that for any single instrument failure, either high or low, the Condition II events itemized in the FSAR Chapter 15 are bounding.

LOSS OF POWER TO AN INVERTER, PROTECTION SET, CONTROL GROUP, OR PROCESS RACK

The NSSS in each Unit of the Watts Bar Nuclear Plant receives power from four inverters (for convenience referred to as inverters I through IV). Each inverter powers a single protection set and a single control group (inverter I powers protection set I and control group 1, inverter II powers protection set II and control group 2, etc.). Therefore, loss of power to one inverter causes a loss of power to both a protection set and a control group.

Besides the loss of an inverter feeding both a protection set and a control group, there is also a possibility of losing power to just a protection set or a control group (for example, through the failure of a fuse or circuit breaker).

In addition to the possibility of losing power to a protection set or control group, there is a chance of losing power to a single process rack (cabinet) in a protection set or control group. Watts Bar has 7100 process gear, and each protection set and control group is comprised of three to four process racks that each have their own breaker and/or fuse.

Table 2 presents the consequences of loss of power to any of the process control racks. For each rack, the table lists the control system affected, the control signal affected, its failure direction, the failure's effect on the control system, and the bounding FSAR Condition II event. Where no control action occurs or where control action is in a safe direction, no bounding event is given.

Tables 3 through 6 give the consequences of loss of power to protection sets I through IV, respectively. Tables 7 through 10 give the analyses for loss of power to control groups 1 through 4, respectively. And, tables 11 through 14 give the analyses for loss of power to inverters I through IV, respectively. The data in all of these tables is presented in the same manner as that for loss of power to a process rack described in the preceding paragraph.

A few of the control system signals analyzed in the loss of power analyses are not completely contained in the NSSS process control racks. Specifically, the four Power Range Flux channels originate in the Nuclear Instrumentation System racks but are powered from inverters I through IV, respectively; the controllers for steamline pressure in the steam dump system are powered from a different inverter than are the corresponding instrument loops; and the condenser available signal for the steam dump system does not go through the process control racks, but is powered from inverter III (power for the pressure switches) and inverter IV (power for the pump running signal). These signals are all appropriately included in the loss of power to inverters I through IV analyses.

Besides the loss of power to an individual process rack, there is the chance of having an electrical fault on one of the control system modules. The control systems are designed so that each module is used in only one control system such that a module failure cannot directly impact more than one control system. A failure in a module would cause the controller to generate either an "off" or a "full on" output, depending on the type of failure. This result would be similar to having a fault in a sensor feeding the control system. Therefore, the failure of or loss of power in any control system module would be covered by the Loss of Any Single Instrument analysis described in Table 1.

The tables show that for a loss of power to any inverter, protection set, control group, or process rack, the Condition II events analyzed in the FSAR Chapter 15 are bounding.

BREAK OF COMMON INSTRUMENT LINES

Table 15, Break Of Common Instrument Lines, considers the scenario whereby an instrument line which supplies more than one signal ruptures, causing faulty sensor readings.

Three sets of sensors used for control are located in common lines:

1) Loop steam flow (protection set II, steam generators 1 and 4, and protection set I, steam generators 2 and 3) and narrow range steam generator level (protection set II, steam generators 1 and 4, and protection set I, steam generators 2 and 3).

2) Pressurizer level and pressurizer pressure (level channel I and pressure channel I, level channel II and pressure channel II, level channel III and pressure channels III and IV).

3) T_{cold} and T_{hot} (any loop)

Not shown on the table since they are not part of the plant control system but are used just for protection are the RCS loop flow transmitters. There are three flow transmitters in each loop, with each transmitter having a common high pressure tap but separate and unique low pressure taps. Therefore, a break at the high pressure flow transmitter tap would result in disabling all three flow transmitters in one loop, resulting in a low flow reading for all three transmitters. This would result in a reactor trip if the plant is above the P-8 setpoint, or an annunciation if it is below P-8.

The only malfunction mode explicitly analyzed was a break in the common instrument line at the tap. Another possibility is to have a complete blockage in the sensor tap, causing the sensor to read a constant (before blockage) value. However, this failure mode is not analyzed since it is really not a credible event. There is no anticipated agent available that would cause a tap blockage. The Reactor Coolant System piping and fittings, and the instrument impulse line tubing are all stainless steel, so no products of corrosion are expected. Also, the water chemistry is of high quality which, along with high temperature operation, precludes the presence of solids in the water and assures the maintenance of the solubility of chemicals in the water. In addition, prior to startup, and during any shutdown as well, it is routine maintenance and servicing practice for instrument lines to be blown down to a canister. Since the buildup of sludge is a slow process, any buildup would be detected during response time testing done during shutdown. Therefore, the hypothesis of the presence of a complete blockage of the sensor tap is not sufficiently credible to warrant its consideration as a design basis.

In the extremely unlikely event that a complete instrument line blockage were to occur, the condition is detectable because the reading would become static (no variations over time). In an unblocked channel, a reading would always vary somewhat due to noise (i.e. flow induced noise in flow channels) or slight controller action (i.e. cycling operation of spray and heaters in pressurizer). By a comparison of the static channel to the redundant unblocked channels, the operator would be informed that a blockage in one channel has occurred.

CONCLUSIONS

Tables 1 through 15 illustrate that a failure of individual sensors, loss of power to individual inverters, protection sets, control groups, or process control racks, or a break in common instrument lines all result in events that are bounded by FSAR Chapter 15 analyses. Therefore, the FSAR adequately bounds the consequences of these fundamental failures.

TABLE 1

LOSS OF ANY SINGLE INSTRUMENT

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
Feedpump Discharge Pressure	1 per plant	———	o Feedwater Control	Lo	FW pump speed increases if in auto mode. (FW control valves close due to increased flow if in auto mode.)	If FW pump in manual - no event. If FW pump and FCV in auto - new steady state w/higher pump speed and decr. FCV lift. If FW pump in auto and FCV in manual - bounding event is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10)
				Hi	FW pump speed decreases if in auto mode. (FW control valves open due to decreased flow if in auto mode).	If FW pump in manual- no event. Other modes -result in a decreased FW flow over time, hence bounding event is loss of Normal FW Flow (FSAR 15.2.8)
Steam Header Pressure	1 per plant	———	o Feedwater Control o Steam Dump (TAVG Mode)	Lo	FW pump speed decreases if in auto mode. (FW control valves open due to decreased flow if in auto mode).	If FW pump in manual - no event. Other modes -result in a decreased FW flow over time, hence bounding event is loss of Normal FW Flow (FSAR 15.2.8)

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
				Hi	FW pump speed increases if in auto mode. (FW control valves close due to decreased flow if in auto mode).	If FW pump in manual - no event. If FW pump and FCV in auto - new steady state w/higher pump speed and decr. FCV lift. If FW pump in auto and FCV in manual - bounding event is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10)
Steam Header Pressure	1 per plant	———	o Feedwater Control o Steam Dump (Pressure Mode)	Lo	FW pump speed decreases if in auto mode. (FW control valves open due to decreased flow if in auto mode).	If FW pump in manual - no event. Other modes -result in a decreased FW flow over time, hence bounding event is loss of Normal FW Flow (FSAR 15.2.8)
				Hi	FW pump speed increases if in auto mode. (FW control valves close due to decreased flow if in auto mode). Dump valves open (Steam dump blocked on Lo-Lo TAVG (P-12)).	Steam dump in pressure mode at hot standby or very low power only. Hence, dump valves will open for only a very short time till lo-lo TAVG (P-12) is reached. If FW pump is in manual or FW pump and FCV in auto, then this event is

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
						bounded by Excessive Load Increase Incident (FSAR 15.2.11). If FW pump in auto and FCV in manual, get increase in FW flow causing excessive cooling. Bounding event is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10)
Loop Steam Flow	2 per loop	1 selected for control	o Feedwater Control	Lo	FW pump speed decreases if in auto mode. FW valves close if in auto mode.	If FW pump and FCV in manual - no event. Other modes result in decreased FW flow, bounding event is Loss of Normal FW Flow (FSAR 15.2.8).
				Hi	FW pump speed increases if in auto mode. FW valves open if in auto mode.	If FW pump and FCV in manual - no event. Other modes - result in increased FW flow, bounding event is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10)

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
Loop FW Flow	2 per loop	1 selected for control	o Feedwater Control	Lo	FW valve opens if in auto mode	If FCV in manual - no event. If FCV in auto, result is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10)
				Hi	FW valve closes if in auto mode	If FCV in manual - no event. If FCV in auto, result is decreased FW flow. Bounding event is Loss of Normal FW Flow (FSAR 15.2.8)
Narrow Range Level	3 per Steam Generator (one available for control)	Control Channel	o Feedwater Control	Lo	FW valve opens if in auto mode	If FCV in manual - no event. If FCV in auto, result is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10)
				Hi	FW valve closes if in auto mode.	If FCV in manual - no event. If FCV in auto, result is decreased FW flow. Bounding event is loss of Normal FW Flow (FSAR 15.2.8)

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
Pressurizer Level (Control)	3 per plant	I or III	o Prz. Level Control	Lo	Charging flow increases. Heaters blocked (except for local control). Letdown isolated (VCT empties, charging pumps take suction from RWST.)	Possible reactivity change bounded by Uncontrolled Boron Dilution (FSAR 15.2.4) Increased inventory enveloped by Inadvertent Operation of ECCS During Power Operation (FSAR 15.2.14)
				Hi	Charging flow decreases Backup heaters on (Later, letdown isolation from interlock channel, heaters blocked from interlock channel.)	While heaters are on, no net depressurization of RCS. After heaters are blocked, decreased charging flow acts to depressurize RCS. Depressurization event is therefore bounded by Accidental Depressurization of the RCS (FSAR 15.2.12)
Pressurizer Level (Interlock)	3 per plant	II or III	o Prz. Level Control	Lo	Letdown isolated. Prz. heaters blocked (except for local control).(Charging flow reduced to maintain level).	Steady-state reached at slightly high level. No event.
				Hi	No control action, get Hi level annunciation.	Not applicable

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
Pressurizer Pressure	4 per plant	I	o Prz. Pressure Control (Pos. 1 or 2)*	Lo	Turn on Htrs. PORV 455A blocked from opening. PORV 456 opens if required, closes when pressure falls below dead band. Spray remains off.	Heaters being on causes increase in Prz. pressure to PORV 456 actuation. No event.
				Hi	Variable htrs. turned off. PORV 455A opens, closes when pressure falls below low pressure interlock. Spray turned on.	Result is bounded by Accidental Depressurization of the RCS (FSAR 15.2.12)
			(Pos. 3)*		Ch. not connected	Not applicable
Pressurizer Pressure	4 per plant	II	o Prz. Pressure Control (Pos. 2 or 3)*	Lo	No control action. PORV 456 blocked from opening. PORV 455A still available for normal control.	Not applicable
				Hi	PORV 456 opens, closes when pressure falls below low pressure interlock. RCS depressurization causes heaters to turn on.	Result is bounded by Accidental Depressurization of the RCS (FSAR 15.2.12)
			(Pos. 1) *		Ch. not connected.	Not applicable

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
Pressurizer Pressure	4 per plant	III	o Prz. Pressure Control (Pos. 3)*	Lo	Turn on Htrs. PORV 455A and 456 blocked from opening. Spray remains off.	Heaters being on causes increase in Prz. pressure, possibly to safety valve actuation. No event.
				Hi	PORV 455A opens, closed on low pressure interlock. Spray turned on. Variable htrs. turned off.	Result is bounded by Accidental Depressurization of the RCS (FSAR 15.2.12)
			(Pos. 1 or 2)*	Lo	Block PORV 456 from opening; no control action. PORV 455A still available for normal control.	Not applicable
				Hi	Unblock PORV 456; no control action.	Not applicable
Pressurizer Pressure	4 per plant	IV	o Prz. Pressure Control (Pos. 1)*	Lo	Block PORV 456 & 455A from opening; no control action	Not applicable
				Hi	PORV 455A unblocked. PORV 456 opens, closes when pressure falls below low pressure interlock. RCS depressurization causes heaters to turn on.	Result is bounded by Accidental Depressurization of the RCS (FSAR 15.2.12)
			(Pos. 2 or 3)*	Lo	Block PORV 455A from opening; no control action	Not applicable

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
				Hi	Unblocked PORV 455A; no control action	Not applicable
TAVG	one per loop	Any Auct. Hi	o Steam Dump (TAVG Mode) o Reactor Control o Prz. Level Control	Lo	No control action	Not applicable
				Hi	Rods in (safe direction). Charging flow increases until full power Prz. level is reached (if at reduced power). If turbine trips, steam dump enabled and dump valves open until steam dump stops when Lo-Lo TAVG is reached.	No event until turbine trips, then dump valves open and bounding event is Excessive Load Increase Incident (FSAR 15.2.11)
TAVG	one per loop	Any Auct. Hi	o Steam Dump (Pressure Mode) o Reactor Control o Prz. Level Control	Lo	No control action.	Not applicable
				Hi	Rods in (safe direction). Charging flow increases until full power Prz. level is reached (if at reduced power). Eventually, reactor trip.	No event.

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
Steamline Pressure	3 per loop for protection, 1 per loop for control (different from those used for protection)	Control Channel	o Steam Dump	Lo	No control action.	Not applicable
				Hi	S. GEN. relief valve opens.	Result is bounded by Accidental Depressurization of the Main Steam System (FSAR 15.2.13)
Turbine Impulse Chamber Pressure	2 per turbine (1 for Control, 1 for Interlock)	Control Channel	o Steam Dump (TAVG Mode) o Reactor Control	Lo	Rods in (safe direction), auto rod withdrawal blocked (C-5). When turbine trip occurs, steam dump unblocked and dump valves modulate until no load TAVG is reached.	Not applicable
				Hi	Rods out until blocked by Hi flux, overpower, or overtemperature rod stop, or until programmed TREF limit is reached. (If turbine trip occurs, steam dump unblocked and dump valves open until no load TAVG is reached).	Result is bounded by Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR 15.2.2)

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
Turbine Impulse Chamber Pressure	2 per turbine	Control Channel	o Steam Dump (Pressure Mode) o Reactor Control	Lo	Rods in (safe direction), auto rod withdrawal blocked (C-5). Steam dump functions normally.	Not applicable
				Hi	Rods out until blocked by Hi flux, overpower, or overtemperature rod stop. Possible reactor trip. Steam dump valves keep steam header pressure at or below setpoint.	Result is bounded by Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR 15.2.2)
Turbine Impulse Chamber Pressure	2 per Turbine	Interlock Channel	o Steam Dump (Tavg Mode)	Lo	Unblock steam dump	Not applicable
				Hi	Block steam dump unless a turbine trip occurs.	Not applicable
Turbine Impulse Chamber Pressure	2 per Turbine	Interlock Channel	o Steam Dump (Pressure Mode)	Lo or Hi	No control action.	Not applicable
Power Range Flux	4 per plant (4 channels used for Reactor Control, 2 of the 4 channels used for FW Control)	either of the 2 used in FW Control	o Reactor Control (auctioneered high) o FW Control	Lo	If main FW valves in auto, FW flow in loops 1 and 4, or in 2 and 3 decreases, and corresponding S.G. levels decrease toward no load values. Unless at low power, reactor trip on low FW flow or low-low level.	Loss of Normal FW Flow (FSAR 15.2.8)

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
				Hi	Auto and manual rod withdrawal blocked (C-2); rods in. If in auto, main FW valves 1 and 4, or 2 and 3 open, and corresponding S.G. levels increase toward full power values (if at reduced power). Rods move in until power mismatch signal decays. TAVG decreases until neg. reactivity from rods is matched by pos. reactivity from cooldown.	Steady-state reached, at same power level but with a lower TAVG. No event.
Power Range Flux	4 per plant	either of the 2 not used in FW Control	o Reactor Control (auctioneered high)	Lo	No control action.	Not applicable
				Hi	Auto and manual rod withdrawal blocked (C-2). Rods move in until power mismatch signal decays. TAVG decreases until neg. reactivity from rods is matched by pos. reactivity from cooldown.	Steady-state reached, at same power level but with a lower TAVG. No event.
Condenser Available	2 per condenser	Any	o Steam Dump	Lo	No control action. Condenser available.	Not applicable
				Hi	No control action-steam dump blocked, condenser unavailable.	Not applicable

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
TAVG High Auctioneer	1 per plant	———	o Steam Dump o Reactor Control o Prz. Level Control	Lo	Steam dump blocked (TAVG mode). Backup htrs. on, charging flow decreased till no-load level reached. Rods out, TAVG and core power increase until blocked by high flux, over-power, or overtemperature rod stop.	Result is bounded by Uncontrolled Rod Cluster Control Assembly Bank With-drawal at Power (FSAR 15.2.2)
				Hi	Indentical to TAVG channel failing high, see analysis above.	See above
Power Range Flux High Auctioneer	1 per plant	———	o Reactor Control	Lo	Rods out, TAVG and core power increase until blocked by high flux, overpower, or overtem-perature rod stop.	Result is bounded by Uncontrolled Rod Cluster Control Assembly Bank With-drawal at Power (FSAR 15.2.2)
				Hi	Identical to either of the 2 Power Range Flux channels not used in FW Control failing high; see analysis above.	See above.
Steamline Pressure Compensator for Steam Flow	2 per loop	1 selected for control	o Steam Flow	Lo	Identical to Loop Steam Flow channel failing low. See analysis above.	See above

SENSOR	NUMBER OF CHANNELS	FAILED CHANNEL	SYSTEM	ASSUMED FAILURE DIRECTION	EFFECT	BOUNDING EVENT
				Hi	Identical to Loop Steam Flow channel failing high. See analysis above.	See above

* Signals for pressurizer pressure and turbine impulse chamber pressure can be obtained from different channels. Selection of desired channels is done by manual switches in the control room. Resulting event due to failed instrument is dependent on switch positions.

TABLE 2

LOSS OF POWER TO RACK 1
(PROTECTION SET I)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Pressure	o Pressurizer Pressure Channel I	Lo	If this signal used for control: Turn on heaters. PORV 455A blocked from opening. PORV 456 opens if required, closes when pressure falls below deadband. Spray remains off. If this signal not used for control, no control action.	Bounding Event is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10). Effects on the pressurizer (possible reactivity changes, increased inventory, increase in pressure to PORV setpoint, or no effect) are not as significant in comparison.
Pressurizer Level	o Pressurizer Level Channel I	Lo	If this signal used for control: Charging flow increases. Heaters blocked (except for local control). Letdown isolated. If this signal not used for control, no control action.	
FW Control	o Narrow Range Level (Control channels for loops 2 and 3)	Lo	If in auto, FW valves in affected loops open. Turbine trip and feedwater isolation on S.G. Hi-Hi water level.	

LOSS OF POWER TO RACK 2

(PROTECTION SET I)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 1)	Lo	No control action.	Not applicable.
Steam Dump	o TAVG (loop 1)	Lo	No control action.	
Pressurizer Level	o TAVG (loop 1)	Lo	No control action.	

LOSS OF POWER TO RACK 3
(PROTECTION SET I)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
FW Control	o Steam Flow (loops 1 and 2) o FW Flow (loops 1 and 2)	Lo Lo	Feedwater flow may increase, decrease, or remain unchanged in the affected loops depending on the position of the channel selector switches. Also, feedwater pump speed may decrease, possibly causing a reduction in flow to the other steam generators.	The bounding event is either Loss of Normal FW Flow (FSAR 15.2.8) or Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10).
FW Pump Speed Control	o Steam Flow (loops 1 and 2)	Lo	See above.	

LOSS OF POWER TO RACK 4
(PROTECTION SET I)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o Turbine Impulse Chamber Pressure (Control channel)	Lo	Rods move in , auto rod withdrawal blocked (C-5).	Bounding Event is either Loss of Normal FW Flow (FSAR 15.2.8) or Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10).
Steam Dump	o Turbine Impulse Chamber Pressure (Control channel)	Lo	If in Pressure Mode, steam dump functions normally. If in TAVG Mode, steam dump unblocked if turbine trip occurs and dump valves modulate until no load TAVG is reached.	
FW Control	o Steam Flow (loops 3 and 4)	Lo	Same effect as in Rack 3.	
	o FW Flow (loops 3 and 4)	Lo		
FW Pump Speed Control	o Steam Flow (loops 3 and 4)	Lo	Same effect as in Rack 3.	

LOSS OF POWER TO RACK 5
(PROTECTION SET II)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Pressure	o Pressurizer Pressure (Channel II)	Lo	If pressure channel selector switch in position 1, no effect. If switch in position 2 or 3, no control action. PORV 456 blocked from opening. PORV 455A still available for normal control.	Bounding Event is Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10).
Pressurizer Level	o Pressure Level (Channel II)	Lo	If level channel selector switch in position 3, no effect. If switch in position 1 or 2, letdown isolated. Pzr. heaters blocked (except for local control). (Charging flow reduced to maintain level).	
FW Control	o Narrow Range Level (Control channels for loops 1 and 4)	Lo	Same effect as in Rack 1.	

LOSS OF POWER TO RACK 6
(PROTECTION SET II)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 2)	Lo	No control action.	Not applicable.
Steam Dump	o TAVG (loop 2)	Lo	No control action.	
Pressurizer Level	o TAVG (loop 2)	Lo	No control action.	

LOSS OF POWER TO RACK 7
(PROTECTION SET II)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
FW Control	o Steam Flow (loops 1 and 3)	Lo	Same effect as in Rack 3.	Same bounding event as in Rack 3.
	o FW Flow (loops 1 and 3)	Lo		
FW Pump Speed Control	o Steam Flow (loops 1 and 3)	Lo	Same effect as in Rack 3.	

LOSS OF POWER TO RACK 8
(PROTECTION SET II)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Steam Dump	o Turbine Impulse Chamber Pressure (Interlock channel)	Lo	If steam dump in Pressure Mode, no effect. If steam dump in TAVG Mode, unblock steam dump. No control action.	Bounding event is either Loss of Normal FW Flow (FSAR 15.2.8) or Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10).
FW Control	o Steam Flow (loops 2 and 4)	Lo	Same effect as in Rack 3.	
	o FW Flow (loops 2 and 4)	Lo		
FW Pump Speed Control	o Steam Flow (loops 2 and 4)	Lo	Same effect as in Rack 3.	

LOSS OF POWER TO RACK 9
(PROTECTION SET III)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Pressure	o Pressurizer Pressure (Channel III)	Lo	If pressure channel selector switch in position 1 or 2, PORV 456 blocked from opening. PORV 455A still available for normal control. No control action. If switch in position 3, turn on heaters. PORV 455A and 456 blocked from opening. Spray remains off.	Combining the effects on the pressurizer pressure and level control systems could result in either increased charging flow and letdown isolation with heaters blocked, or else heaters on and spray off with PORV's blocked. The latter causes an increase in pressure, possibly to the safety valve setpoint. The former result is bounded by two events - a possible reactivity change is bounded by Uncontrolled Boron Dilution (FSAR 15.2.4), and increase in inventory is enveloped by Inadvertent Operation of ECCS During Power Operation (FSAR 15.2.14).
Pressurizer Level	o Pressurizer Level (Channel III)	Lo	If level channel selector switch in position 2, no effect. If switch in position 1, charging flow increases. Heaters blocked (except for local control). Letdown isolated. If switch in position 3, letdown isolated. Heaters blocked (except for local control). (Charging flow reduced to maintain level).	

LOSS OF POWER TO RACK 10
(PROTECTION SET III)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 3)	Lo	No control action.	Not applicable.
Steam Dump	o TAVG (loop 3)	Lo	No control action.	
Pressurizer Level	o TAVG (loop 3)	Lo	No control action.	

LOSS OF POWER TO RACK 11
(PROTECTION SET III)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
None of the major control systems affected.				Not applicable.

LOSS OF POWER TO RACK 12
(PROTECTION SET IV)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
None of the major control systems affected.				Not applicable.

LOSS OF POWER TO RACK 13
(PROTECTION SET IV)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 4)	Lo	No control action.	Not applicable.
Steam Dump	o TAVG (loop 4)	Lo	No control action.	
Pressurizer Level	o TAVG (loop 4)	Lo	No control action.	

LOSS OF POWER TO RACK 14
(Control Group 1)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
None of the major control systems affected.				Not applicable.

LOSS OF POWER TO RACK 15

(Control Group 1)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Pressure	o Pressurizer Pressure (Channels I and III)	Closed/Off	PORV 455A blocked by control, PORV 456 blocked by interlock. Spray valves close, variable htrs. off.	Bounding event is Loss of Normal FW (FSAR 15.2.8).
Pressurizer Level	o Pressurizer Level (Channels II and III)	Off	Heater actuation block via interlock channel disabled, letdown isolation by valve 460 blocked.	
FW Control (loop 1)	o All (System Deenergized)	Closed/Off	Main FW valve 1 closes. Loss of feedwater to S.G. 1. Reactor trip on low feedwater flow or low-low water level in S.G. 1.	

LOSS OF POWER TO RACK 16

(Control Group 1)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Steam Dump	o TAVG (High Auctioneered) o Turbine Impulse Chamber Pressure (Control and Interlock Channels) o S.G. Header Pressure	Closed/Off	No control action. Steam Dump unavailable in either Mode.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
FW Control	o Power Range Flux (loop 1)	Lo	If main FW valve 1 in auto, FW flow in loop 1 decreases and S.G. 1 level decreases toward no load value. Unless at low power, reactor trip on low feedwater flow or low-low water level in S.G. 1.	
FW Pump Speed Control	o All (System Deenergized)	Off	Feedwater pump speed decreases (if in auto). Feedwater flow in loop 1 decreases, and FW flow in the other loops may decrease depending on position of the control valves.	

LOSS OF POWER TO RACK 17
(Control Group 2)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
None of the major control systems affected.				Not applicable.

LOSS OF POWER TO RACK 18
(Control Group 2)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o Turbine Impulse Chamber Pressure (Control Channel)	Lo	Rods move in (safe direction).	Possible reactivity change bounded by uncontrolled Boron Dilution (FSAR 15.2.4). Increased inventory enveloped by Inadvertent Operation of ECCS during power operation (FSAR 15.2.14).
Steam Dump	o Turbine Impulse Chamber Pressure (Control Channel)	Lo	In TAVG Mode, steam dump demanded but blocked by other signals. If turbine trip occurs, steam dump unblocked and dump valves modulate until no load TAVG is reached. In Pressure Mode, steam dump functions normally.	
Pressurizer Level	o TAVG (High Auctioneered) o Pressurizer Level (Channels I and III)	Open/Speedup	Letdown isolation by valve 459 blocked. Charging flow control valve fails open. PD charging pump goes to max. speed. RCS inventory increases; in time, reactor trip on high pressurizer level.	

LOSS OF POWER TO RACK 19
(Control Group 2)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Pressure	o Pressurizer Pressure (Channels II and IV)	Closed	PORV 456 blocked by control, PORV 455A blocked by interlock. No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
FW Control (loop 2)	o All (System Deenergized)	Closed/Off	Main FW valve 2 closes. Loss of feedwater to S.G. 2. Reactor trip on low feedwater flow or low-low water level in S.G. 2.	

LOSS OF POWER TO RACK 20
(Control Group 3)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
None of the major control systems affected.				Not applicable.

LOSS OF POWER TO RACK 21
(Control Group 3)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Steam Dump	o Steamline Pressure (Control Channels for loops 2 and 3)	Lo/Closed	Relief valves in loops 2 and 3 unavailable. No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
FW Control (loop 3)	o All (System Deenergized)	Closed/Off	Main FW valve 3 closes. Loss of feedwater to S.G. 3. Reactor trip on low feedwater flow or low-low water level in S.G. 3.	

LOSS OF POWER TO RACK 22

(Control Group 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
None of the major control systems affected.				Not applicable.

LOSS OF POWER TO RACK 23
(Control Group 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
FW Control (loop 4)	o All (System Deenergized)	Closed/Off	Main FW valve 4 closes. Loss of feedwater to S.G. 4. Reactor trip on low feedwater flow or low-low water level in S.G. 4.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).

LOSS OF POWER TO RACK 24
(Control Group 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (High Auctioneered)	Lo	Steam dump blocked (TAVG Mode). Backup htrs. on, charging flow decreased till no-load level reached. Rods out, TAVG and core power increase until blocked by high flux, over-power, or overtemperature rod stop.	Bounding Event is Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR 15.2.2).
Pressurizer Level	o TAVG (High Auctioneered)	Lo		
Steam Dump	o TAVG (High Auctioneered)	Lo		
	o Steamline Pressure (Control Channels for loops 1 and 4)	Lo/Closed	Relief valves in loops 1 and 4 unavailable. No control action.	

TABLE 11

LOSS OF POWER TO INVERTER I
(PROTECTION SET I AND CONTROL GROUP 1)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 1)	Lo	Rods move in, auto rod withdrawal blocked (C-5).	Bounding Event is Loss of Normal FW (FSAR 15.2.8). Effects on the pressurizer are not as significant in comparison.
	o Turbine Impulse Chamber Pressure (Control Channel)	Lo		
	o Power Range Flux	Lo		
Steam Dump	o TAVG (loop 1 and High Auctioneered)	Closed/Off	Steam Dump unavailable in either Mode. Relief valves in loops 1 and 4 unavailable. No control action.	
	o Turbine Impulse Chamber Pressure (Control and Interlock Channels)			
	o S.G. Header Pressure			
	o Steamline Pressure Controllers (loops 1 and 4)			

TABLE 11 (Continued)

LOSS OF POWER TO INVERTER I
(PROTECTION SET I AND CONTROL GROUP 1)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Pressure	o Pressurizer Pressure (Channels I and III)	Closed/Off	Both PORV's blocked. Spray valves close, variable htrs. off.	
Pressurizer Level	o TAVG (loop 1)	Lo	Heater actuation block via interlock channel disabled. Letdown isolation by valve 460 blocked.	
	o Pressurizer Level (Channel I)	Lo	If level channel selector switch in position 2 or 3, charging flow increases. Heaters blocked (except for local control). Letdown isolation by valve 459.	
	(Channels II and III)	Off		
FW Control	o All - loop 1 (System Deenergized)	Closed/Off	Main FW valve 1 closes. Loss of feedwater to S.G. 1. FW flow in loops 2 and 3 increases. FW flow in loop 4 decreases. Reactor trip on low feedwater flow or low-low water level in S.G. 1.	
	o Steam Flow (all other loops)	Lo		
	o FW Flow (all other loops)	Lo		

TABLE 11 (Continued)

LOSS OF POWER TO INVERTER I
(PROTECTION SET I AND CONTROL GROUP 1)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
	o Narrow Range Level (Control Channels for loops 2 and 3)	Lo		
	o Power Range Flux (affecting loops 1 and 4)	Lo		
FW Pump Speed Control	o All (System Deenergized)	Off	Feedwater pump speed decreases (if in auto), causing a reduction in flow in loops 2, 3, and 4. (FW flow in loop 1 has already been cut off - see above).	

TABLE 12

LOSS OF POWER TO INVERTER II
(PROTECTION SET II AND CONTROL GROUP 2)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 2)	Lo	Rods move in (safe direction).	Bounding Event is Loss of Normal FW (FSAR 15.2.8). Effects on the pressurizer are not as significant in comparison.
	o Turbine Impulse Chamber Pressure (Control Channel)	Lo		
	o Power Range Flux	Lo		
Steam Dump	o TAVG (loop 2)	Lo	In Pressure Mode, steam dump functions normally. If in TAVG Mode, steam dump unblocked and dump valves modulate until no load TAVG is reached.	
	o Turbine Impulse Chamber Pressure (Control and Interlock Channels)	Lo		
Pressurizer ·sure	o Pressurizer Pressure (Channels II and IV)	Closed	PORV's 456 and 455A both blocked. No control action.	
Pressurizer Level	o TAVG (loop 2 and High Auctioneered)	Open/Speedup Closed/Blocked	Letdown isolation by valve 459 blocked. Charging flow control valve fails open. PD charging pump goes to max. speed. If level channel selector switch in position 1 or 2, letdown isolation by valve 460.Htrs. blocked (except for local control).	
	o Pressurizer Level (Channels I, II, and III)			

TABLE 12 (Continued)

LOSS OF POWER TO INVERTER II
(PROTECTION SET II AND CONTROL GROUP 2)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
FW Control	o All - loop 2 (System Deenergized)	Closed/Off	Main FW valve 2 closes. Loss of feedwater to S.G. 2. FW flow in loops 1 and 4 increases. FW flow in loop 3 decreases. Reactor trip on low feedwater flow or low-low water level in S.G. 2.	
	o Steam Flow (all other loops)	Lo		
	o FW Flow (all other loops)	Lo		
	o Narrow Range Level (Control Channels for loops 1 and 4)	Lo		
	o Power Range Flux (affecting loops 2 and 3)	Lo		
FW Pump Speed Control	o Steam Flow (all loops)	Lo	Depending on the position of the channel selector switches, feedwater pump speed may decrease, possibly causing a reduction in flow to loops 1, 3, and 4. (FW flow in loop 2 has already been cut off - see above).	

TABLE 13

LOSS OF POWER TO INVERTER III
(PROTECTION SET III AND CONTROL GROUP 3)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 3)	Lo	No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8). Effects on the pressurizer are not as significant in comparison.
	o Power Range Flux	Lo		
Steam Dump	o TAVG (loop 3)	Lo	Steam dump blocked (condenser unavailable). Relief valves in loops 2 and 3 unavailable. No control action.	
	o Steamline Pressure (Control Channels for loops 2 and 3)	Lo/Closed		
	o Condenser Available (Pressure Switches)	Off		
Pressurizer Pressure	o Pressurizer Pressure (Channel III)	Lo	If pressure channel selector switch in position 1 or 2, PORV 456 blocked from opening. PORV 455A still available for normal control. No control action. If switch in position 3, turn on heaters. PORV's 455A and 456 blocked from opening. Spray remains off.	

TABLE 13 (Continued)

LOSS OF POWER TO INVERTER III
(PROTECTION SET III AND CONTROL GROUP 3)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Level	o TAVG (loop 3) o Pressurizer Level (Channel III)	Lo Lo	If level channel selector switch in position 2, no effect. If switch in position 1, charging flow increases. Heaters blocked (except for local control). Letdown isolated. If switch in position 3, letdown isolated. Heaters blocked (except for local control). (Charging flow reduced to maintain level).	
FW Control (loop 3)	o All (System Deenergized)	Closed/Off	Main FW valve 3 closes. Loss of feedwater to S.G. 3. Reactor trip on low feedwater flow or low-low water level in S.G. 3.	

TABLE 14

LOSS OF POWER TO INVERTER IV
(PROTECTION SET IV AND CONTROL GROUP 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o All (System Deenergized)	Off	Rods remain stationary (not available for control). No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
Steam Dump	o TAVG (loop 4 and High Auctioneered)	Lo	Steam dump blocked. Relief valves in all 4 loops unavailable. No control action.	
	o Steamline Pressure (Control Channels for loops 1 and 4)	Lo/Closed		
	o Steamline Pressure Controllers (loops 2 and 3)	Off		
	o Condenser Available (Pump Running Signal)	Off		
Pressurizer Pressure	o Pressurizer Pressure (Channel IV)	Lo	If pressure channel selector switch in position 2 or 3, PORV 455A blocked from opening. No control action. If switch in position 1, PORV's 456 and 455A both blocked from opening. No control action.	

TABLE 14 (Continued)

LOSS OF POWER TO INVERTER IV
(PROTECTION SET IV AND CONTROL GROUP 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Level	o TAVG (loop 4 and High Auctioneered)	Lo	Backup htrs. on, charging flow decreased till no load level reached.	
FW Control	o All - loop 4 (System Deenergized)	Closed/Off	Main FW valve 4 closes. Loss of feedwater to S.G. 4. If other main FW valves in auto, FW flow in other loops decreases toward no load levels. Reactor trip on low feedwater flow or low-low water level in S.G. 4.	
	o Power Range Flux (all other loops)	Lo		

TABLE 15

LOSS OF COMMON INSTRUMENT LINES

(ASSUMED BREAK IN LINE)

Sensors	Failed Channels	System	Failure Direction	Effect	Bounding Event
Loop Steam Flow and Narrow Range Level	I or II	Feedwater Control	Lo Hi	FW valve closes in affected S.G., pump speed decreases.	Bounding event is Loss of Normal FW (FSAR 15.2.8).
Pressurizer Level (Control) and Pressurizer Pressure (PORV 455A)	I (Level and Pressure)	Prz. Level Control Prz. Pressure Control	Hi Lo	PORV 455A blocked. Spray off. Charging flow decreases (Control). Backup heaters on (Control). (On low level, letdown isolated and heaters blocked from interlock channel).	After htrs. are blocked, decreased charging flow acts to depressurize RCS, which is bounded by Accidental Depressurization of the RCS (FSAR 15.2.12.).
Pressurizer Level (Interlock) Pressurizer Pressure (PORV 456)	II (Level and Pressure)	Prz. Level Control Prz. Pressure Control	Hi Lo	No control action. PORV 456 blocked. PORV 455A still available for normal control.	Not applicable.
Pressurizer Level (Control or Interlock) and Pressurizer Pressure (Either PORV)	III (Level) III and IV (Pressure)	Prz. Level Control Prz. Pressure Control	Hi Lo	PORV's 455A and 456 blocked. Spray unavailable if on pressure Channel III. Charging flow decreases and backup heaters on if on level control channel. (Later, on low level,	Depending on level switch positions, event is at most a depressurization event which is bounded by Accidental Depressurization of the RCS (FSAR 15.2.12).

TABLE 15 (Continued)

LOSS OF COMMON INSTRUMENT LINES

(ASSUMED BREAK IN LINE)

Sensors	Failed Channels	System	Failure Direction	Effect	Bounding Event
				letdown isolated and heaters blocked from (non-failed) interlock channel. No level control action if on level interlock channel.	
T_{cold} and/or T_{hot}	I, II, III, or IV	Reactor Control Steam Dump Pressurizer Level	Lo or Hi	See failure of TAVG in "Loss of Any Single Instrument", Table 1.	

LOSS OF POWER TO RACK 25
(Control Group 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o All (System Deenergized)	Off	Rods remain stationary. No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
FW Control	o Power Range Flux (loops 1 through 4)	Lo	If main FW valves in auto, FW flow in all 4 loops decreases, and S.G. levels decrease toward no load values. Unless at low power, reactor trip on low FW flow or low-low S.G. water level.	

LOSS OF POWER TO RACK 28
(PROTECTION SET IV)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Pressure	o Pressurizer Pressure (Channel IV)	Lo	If pressure channel selector switch in position 2 or 3, PORV 455A blocked from opening. No control action. If switch in position 1, PORV's 456 and 455A both blocked from opening. No control action.	Not applicable.

TABLE 3

LOSS OF POWER TO PROTECTION SET I
(PROCESS RACKS 1, 2, 3, 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 1)	Lo	Rods move in, auto rod withdrawal blocked (C-5).	Bounding Event is either Loss of Normal FW (FSAR 15.2.8) or Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10). Effects on the pressurizer (possible reactivity changes, increased inventory, increase in pressure to PORV setpoint, or no effect) are not as significant in comparison.
	o Turbine Impulse Chamber Pressure (Control Channel)	Lo		
Steam Dump	o TAVG (loop 1)	Lo	If in Pressure Mode, steam dump functions normally. If in TAVG Mode, steam dump unblocked if turbine trip occurs and dump valves modulate until no load TAVG is reached.	
	o Turbine Impulse Chamber Pressure (Control Channel)	Lo		
Pressurizer Pressure	o Pressurizer Pressure (Channel I)	Lo	If pressure channel selector switch in position 3, no effect. If switch in position 1 or 2, turn on heaters. PORV 455A blocked from opening. PORV 456 opens if required, closes when pressure falls below deadband. Spray remains off.	

TABLE 3 (Continued)

LOSS OF POWER TO PROTECTION SET I
(PROCESS RACKS 1, 2, 3, 4)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Pressurizer Level	o TAVG (loop 1)	Lo	If level channel selector switch in position 1, no effect. If switch in position 2 or 3, charging flow increases.	
	o Pressurizer Level (Channel I)	Lo	Heaters blocked (except for local control). Letdown isolated.	
FW Control	o Steam Flow (all loops)	Lo	Feedwater flow in loops 2 and 3 increases. Feedwater flow in loops 1 and 4 may increase,	
	o FW Flow (all loops)	Lo	decrease, or remain unchanged depending on the position of the channel selector switches.	
	o Narrow Range Level (Control Channels for loops 2 and 3)	Lo	Also, feedwater pump speed may decrease, possibly causing a reduction in flow to the steam generators.	
FW Pump Speed Control	o Steam Flow (all loops)	Lo	See above.	

TABLE 4

LOSS OF POWER TO PROTECTION SET II
(PROCESS RACKS 5, 6, 7, 8)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 2)	Lo	No control action.	Bounding Event is either Excessive Heat Removal Due to FW System Malfunctions (FSAR 15.2.10) or Loss of Normal FW (FSAR 15.2.8).
Steam Dump	o TAVG (loop 2)	Lo	If steam dump in Pressure Mode, no effect. If steam dump in TAVG Mode, unblock steam dump. No control action.	
	o Turbine Impulse Chamber Pressure (Interlock Channel)	Lo		
Pressurizer Pressure	o Pressurizer Pressure (Channel II)	Lo	If pressure channel selector switch in position 1, no effect. If switch in position 2 or 3, PORV 456 blocked from opening. PORV 455A still available for normal control. No control action.	
Pressurizer Level	o TAVG (loop 2)	Lo	If level channel selector switch in position 3, no effect. If switch in position 1 or 2, let-down isolated. Pzr. heaters blocked (except for local control). (Charging flow reduced to maintain level). Steady-state reached at slightly high level.	
	o Pressurizer Level (Channel II)	Lo		

TABLE 4 (Continued)

LOSS OF POWER TO PROTECTION SET II
(PROCESS RACKS 5, 6, 7, 8)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
FW Control	o Steam Flow (all loops)	Lo	Feedwater flow in loops 1 and 4 increases. Feedwater flow in loops 2 and 3 may increase, decrease, or remain unchanged depending on the position of the channel selector switches. Also, feedwater pump speed may decrease, possibly causing a reduction in flow to the steam generators.	
	o FW Flow (all loops)	Lo		
	o Narrow Range Level (Control Channels for loops 1 and 4)	Lo		
FW Pump Speed Control	o Steam Flow (all loops)	Lo	See above.	

TABLE 5

LOSS OF POWER TO PROTECTION SET III
(PROCESS RACKS 9, 10, 11)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 3)	Lo	No control action.	Combining the effects on the pressurizer pressure and level control systems could result in either increased charging flow and letdown isolation with heaters blocked, or else heaters on and spray off with PORV's blocked. The latter result causes an increase in pressure, possibly to the safety valve setpoint. The former result is bounded by two events - a possible reactivity change is bounded by Uncontrolled Boron Dilution (FSAR 15.2.4), and increase in inventory is enveloped by Inadvertent Operation of ECCS During Power Operation (FSAR 15.2.14).
Steam Dump	o TAVG (loop 3)	Lo	No control action.	
Pressurizer Pressure	o Pressurizer Pressure (Channel III)	Lo	If pressure channel selector switch in position 1 or 2, PORV 456 blocked from opening. PORV 455A still available for normal control. No control action. If switch in position 3, turn on heaters. PORV's 455A and 456 blocked from opening. Spray remains off.	
Pressurizer Level	o TAVG (loop 3)	Lo	If level channel selector switch in position 2, no effect. If switch in position 1, charging flow increases. Heaters blocked (except for local control). Letdown isolated. If switch in position 3, letdown isolated. Heaters blocked (except for local control). (Charging flow reduced to maintain level).	
	o Pressurizer Level (Channel III)	Lo		

TABLE 6

LOSS OF POWER TO PROTECTION SET IV
(PROCESS RACKS 12, 13, 28)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o TAVG (loop 4)	Lo	No control action.	Not applicable (no event).
Steam Dump	o TAVG (loop 4)	Lo	No control action.	
Pressurizer Pressure	o Pressurizer Pressure (Channel IV)	Lo	If pressure channel selector switch in position 2 or 3, PORV 455A blocked from opening. No control action. If switch in position 1, PORV's 456 and 455A both blocked from opening. No control action.	
Pressurizer Level	o TAVG (loop 4)	Lo	No control action.	

TABLE 7

LOSS OF POWER TO CONTROL GROUP 1
(PROCESS RACKS 14, 15, 16)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Steam Dump	o TAVG (High Auctioneered)	Closed/Off	Steam Dump unavailable in either Mode. No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
	o Turbine Impulse Chamber Pressure (Control and Inter-lock Channels)			
	o S.G. Header Pressure			
Pressurizer Pressure	o Pressurizer Pressure (Channels I and III)	Closed/Off	PORV 455A blocked by control, PORV 456 blocked by interlock. Spray valves close, variable htrs. off.	
Pressurizer Level	o Pressurizer Level (Channels II and III)	Off	Heater actuation block via interlock channel disabled, letdown isolation by valve 460 blocked.	
FW Control (loop 1)	o All (System Deenergized)	Closed/Off	Main FW valve 1 closes. Loss of feedwater to S.G. 1. Reactor trip on low feedwater flow or low-low water level in S.G. 1.	

TABLE 7 (Continued)

LOSS OF POWER TO CONTROL GROUP 1
(PROCESS RACKS 14, 15, 16)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
FW Pump Speed Control	o All (System Deenergized)	Off	Feedwater pump speed decreases (if in auto). FW flow in loops 2, 3 and 4 may decrease, depending on position of the control valves. (FW flow in loop 1 has already been cut off - see above).	

TABLE 8

LOSS OF POWER TO CONTROL GROUP 2
(PROCESS RACKS 17, 18, 19)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o Turbine Impulse Chamber Pressure (Control Channel)	Lo	Rods move in (safe direction).	Bounding Event is Loss of Normal FW (FSAR 15.2.8). Effects on the pressurizer not as significant in comparison.
Steam Dump	o Turbine Impulse Chamber Pressure (Control Channel)	Lo	In TAVG Mode, steam dump demanded but blocked by other signals. If turbine trip occurs, steam dump unblocked and dump valves modulate until no load TAVG is reached. In Pressure Mode, steam dump functions normally.	
Pressurizer Pressure	o Pressurizer Pressure (Channels II and IV)	Closed	PORV's 456 and 455A both blocked. No control action.	
Pressurizer Level	o TAVG (High Auctioneered) o Pressurizer Level (Channels I and III)	Open/Speedup	Letdown isolation by valve 459 blocked. Charging flow control valve fails open. PD charging pump goes to max. speed. RCS inventory increases.	
FW Control (loop 2)	o All (System Deenergized)	Closed/Off	Main FW valve 2 closes. Loss of feedwater to S.G. 2. Reactor trip on low feedwater flow or low-low water level in S.G. 2.	

TABLE 9

LOSS OF POWER TO CONTROL GROUP 3
(PROCESS RACKS 20, 21)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Steam Dump	o Steamline Pressure (Control Channels for loops 2 and 3)	Lo/Closed	Relief valves in loops 2 and 3 unavailable. No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
FW Control (loop 3)	o All (System Deenergized)	Closed/Off	Main FW valve 3 closes. Loss of feedwater to S.G. 3. Reactor trip on low feedwater flow or low-low water level in S.G. 3.	

TABLE 10

LOSS OF POWER TO CONTROL GROUP 4
(PROCESS RACKS 22, 23, 24, 25)

Control System Affected	Signal Affected	Failure Direction	Itemized Effect	Bounding Event
Reactor Control	o All (System Deenergized)	Off	Rods remain stationary (not available for control). No control action.	Bounding Event is Loss of Normal FW (FSAR 15.2.8).
Steam Dump	o TAVG (High Auctioneered)	Lo	Steam dump blocked (TAVG Mode). Relief valves in loops 1 and 4 unavailable. No control action.	
	o Steamline Pressure (Control Channels for loops 1 and 4)	Lo/Closed		
Pressurizer Level	o TAVG (High Auctioneered)	Lo	Backup htrs. on, charging flow decreased till no-load level reached.	
FW Control	o All - loop 4 (System Deenergized)	Closed/Off	Main FW valve 4 closes. Loss of feedwater to S.G. 4. If other main FW valves in auto, FW flow in other loops decreases toward no load levels. Reactor trip on low feedwater flow or low-low water level in S.G. 4.	
	o Power Range Flux (All other loops)	Lo		

413.11 Question:

Your responses to several parts of question 413.01 are inadequate. Modify the following test descriptions as indicated:

1. Part a, Item A.2.a. You have misinterpreted the recommended test. The test should include testing of the CVCS system's ability to blend concentrated boric acid for injection, verification of sampling paths, and holdup or delay times in sample lines, verification of sampling procedures, and adequacy of heat tracing on concentrated boric acid systems. It should also verify injection and letdown flowpaths and flow rates.

2. Part a, Item A.3. As described in the FSAR, test W7.1 will not verify total channel response time. The response times of hardware between the measured variable and the input to the sensors (i.e., snubbers, flow limiting devices, sensing lines, etc.) have not been accounted for in the test summary or acceptance criteria. Modify the test description to include response time between the measured variable and the sensor. (The delay times of instruments may be accounted for analytically.)

3. Part a, Item A.4.a. Although the test descriptions referenced in your reply do include expansion and restraint tests, they do not include operability tests of pumps, valves, controls, logics, isolations, and interlocks.

4. Part a, Item A.5.r. Expand test TVA-52 to include testing of alarm and recorder functions. Explain what the phrase 'to ensure the assurance and operability' means.

5. Part a, Item A.6.b. Test descriptions TVA-13A, 13B, 13C, 15 and 16 require full load testing only of the emergency diesel generators and the 125VDC batteries. Modify these descriptions or provide additional test descriptions that address the testing of vital buses at rated load from preferred offsite supplies, 125VDC buses from the battery chargers, and the instrument and control buses from normal and alternate power supplies. Tests should include full load tests with normal operating loads, as well as with accident loads.

6. Part a, Item A.10.c. Test description W10.1 addresses the cask loading pit gate and transfer

canal gate in the prerequisites. Describe the testing to determine operability and leak tightness of these sectionalizing devices.

7. Part a, Item A.12.c. Even though the sampling equipment is purchased as a unit, transportation and installation may have damaged some equipment or invalidated the manufacturer's calibration. Describe the preoperational testing to be conducted to demonstrate that the equipment can perform its function within the required accuracy.

8. Part b(4). Test W3.1 does not include provisions for testing the RWST temperature and level indication. Revise the test description to include this testing. Your response to the question on RWST heater testing is unacceptable. Meeting technical specification limits implies more than an economic necessity. Considering the history of failures of vents, reliefs, and isolation valves in concentrated boric acid systems due to precipitation of boric acid, the ability of the heaters to maintain RWST temperature should be tested.

9. Parts b(5) and (6). Expand test descriptions W6.1 and W6.2 to identify what load tests will be conducted on the bridge and crane.

10. Part b(18). Contrary to your response, the individual test descriptions for CVCS, SIS and auxiliaries, ERCW, and CCW do not contain tests of intersystem leakage. Revise these test abstracts, as necessary, or provide a new description for intersystem leakage detection testing.

Response:

1. Part a, Item A.2.a - TVA will provide response at a later date.

2. Part a, Item A.3 - TVA will provide a response at a later date.

3. Part a, Item A.4.a

   The operability test of pumps, valves, controls, logic, isolations, and interlocks for the Steam and Feedwater Systems (where applicable), is described in the test abstracts TVA-38 'Main Feedwater System' and TVA-40 'Main Steam System.'

4. Part a, Item A.5.r

   See revised FSAR Table 14.2-1 test objectives for test TVA-52.

5. Part a, Item A.6.b

   See revised FSAR Table 14.2-1 test objectives for tests TVA-13A, -13B, -13C, 15, and 16.

6. Part a, Item A.10.c

   Testing to determine operability and leak tightness of the cask loading pit gate and transfer canal gate. Items and sequence of testing:

   1. Control air is available to the gate seals.

   2. The transfer canal and cask loading gate seal relief valve, two each, have their setpoints properly verified.

   3. Gate slots and seals are thoroughly cleaned.

   4. Gate seals are inspected for manufacturing defects, alignment and leak tested.

   5. Gates are removed from their storage location and set in place.

   6. Seals are inflated.

   7. Water level is raised in the pit until it is at the high water level of the pit.

   8. During the filling process, all possible areas for possible leakage are observed.

   9. The pit is let stand for 72 hours and observance is continued.

   10. The water level is lowered in the pit, the seals are deflated, and the gates are removed and placed in their storage location.

   See revised FSAR table 14.2-1 test objectives for test W-10.1.

7. Part a, Item A.12.c

   We do not concur with the staff position that a preoperational test is necessary or beneficial in demonstrating functional performance capabilities for laboratory monitoring and analysis equipment. When Radiochemical or Health Physics laboratory equipment is received, it is initially tested in accordance with written plant instructions. These tests provide a functional demonstration that each individual piece of laboratory equipment can perform its intended function within manufacturer's or purchase specification limits. Manufacturer's factory calibration is not used as verification that equipment performance is satisfactory.

Testing is performed using standards traceable to NBS standards and each test is documented. Additional testing of this equipment is performed prior to and periodically after issuance of an operating license to certify that proper calibration is maintained.

We believe that the current identified plant instructions require and document an adequate test of laboratory equipment. For these reasons, a test added to and identified in the preoperational test program would not add any additional measure of assurance that the equipment functions as designed.

8. Part b (4)

RWST temperature and level indicators are tested in W-3.1E and W-7.3. W-3.1E demonstrates that the RWST heaters will be energized upon actuation of the proper temperature switches and verifies high range level instrumentation is used to initiate automatic switchover to recirculation mode in W-7.3.

9. Parts b (5) and (6)

See revised FSAR table 14.2-1 test objectives for test W-6.2.

10. Part b (18) - TVA will provide a response at a later date.

413.12 Question:

Your responses to several parts of question 413.02 are inadequate. Modify the following test descriptions as indicated:

1. Part a. Describe how the operability of the pressurizer relief valves will be demonstrated. Provide acceptance criteria for pressurizer heaters, relief valves, spray valves, PORV's and steam generator atmospheric dumps. Ensure that the acceptance criteria include minimum and maximum flow rates for pressurizer code safeties and PROV's.

2. Part b. Provide acceptance criteria in W1.2 for the functional check of RCS components, pressurizer relief valve response time and operation, relief tank temperature, and the 240 hr. RCS full flow test.

3. Part c. No acceptance criteria have been provided in W1.3 as requested. Provide these acceptance criteria.

4. Part e. Clarify the prerequisites and test summary in W1.8 to identify the testing that will be done on the slowest measured control rod. Provide technical justification for the tests.

5. Part h. Modify acceptance criteria (5) of test W2.2 to specify what 'proper liquid temperatures' are.

6. Part i. No acceptance criteria have been provided in W2.1 as requested. Provide these acceptance criteria.

7. Part j. Verify that test objective (5) of W3.1 will be demonstrated in accordance with your response to question 212.74. Modify the test objective as necessary to reflect this testing. In the test description for the cold leg accumulator instrumentation, identify the 'applicable TVA documentation' which will serve as acceptance criteria. In the UHI test objective, clarify what is meant by the 'performance evaluation.' Provide acceptance criteria for the Recirculation Flow Test in W3.3. Additional information may be required as a result of evaluation of the responses to questions 212.86, 212.79, 212.16, 212.33, and 212.36. Staff review of the testing of ECCS systems will remain an open item until this evaluation is completed.

8. Part k. The response to this question is inadequate. Test description W7.1 does not contain the total response time testing indicated in the original question, and the acceptance criteria from the original test description have been deleted. Provide a description of testing consistent with the original question and complete acceptance criteria consistent with these tests.

9. Part m. Test description W7.3 is not an adequate response. Expand the test description to include response times and acceptance criteria for response times. Expand the acceptance criteria provided to specifically address the test objective and test summary.

10. Part w. This will remain an open item of review until the test description for W9.8 is provided.

11. Part y. Test description W10.9 states that acceptance criteria are provided in individual test sections. Modify the test description to include these acceptance criteria. Also, the response on measuring door opening forces is inadequate. Provide a test summary and acceptance criteria for this testing.

12. Part cc. It is the staff's position that the ventilation system tests described in TVA-4, 5, 6, and 7 should be conducted at hot operating conditions and extrapolated to design conditions. Test data should confirm the adequacy of the as build systems to maintain temperatures within design values. Revise the test descriptions to conform to this position. Provide acceptance criteria for the tests and specify what 'applicable flow diagrams' are.

13. Part ee. It is the staff's position that equipment in spaces be operated to simulate worst case accident heat loads. Sufficient test data should be taken to extrapolate the heating or cooling capability of HVAC systems to worst-case accident conditions. Revise test descriptions TVA-9A, 9B, and 9C to reflect this position. Provide acceptance criteria for test TVA-9C. Define 'applicable logic and flow diagrams,' as used in TVA-9C.

14. Part ff. Revise the test description to include testing of filters and absorbers in accordance with Regulatory Guide 1.52 and to include leak tightness testing. Provide acceptance criteria for these

tests.

15. Part ii. Revise test TVA-13C to conform to Regulatory Guide 1.108 as indicated in your reply to question 040.69. Expand the test description to identify the status of the plant's preferred electrical distribution systems during the test. Describe how manual transfer devices for loads supplied by two redundant divisions will be demonstrated.

16. Part mm. Contrary to your response, there is no test description for TVA-17. Provide this test description.

17. Part nn. In test TVA-18, Section (6), identify the 'various FSAR sections' which contain the acceptance criteria or summarize the criteria in the test description.

18. Part oo. Test description TVA-20 fails to describe how system redundancy will be demonstrated as requested. Modify the description to include this testing.

19. Part pp. In test description TVA-22, verify that the turbine driven pump will be started from cold conditions, i.e., the pump and pump auxiliaries are at essentially ambient conditions and no system priming, preheating, or preparation which would not be done during normal operation has been done. Provide acceptance criteria for rated flow at various steam generator pressures and for successful operation in the recirc mode.

20. Part qq. Provide test acceptance criteria for each described test in TVA-25.

21. Part rr. Section 9.3.1.4 of your FSAR commits to a test of the compressed air system in accordance with Regulatory Guide 1.80. Expand test description TVA-27 to comply with that commitment. It is the staff's position that testing should be performed on service air, control air, and auxiliary control air to verify that the systems operate in accordance with design and that no unanticipated failure modes would result in a transient more severe than those considered in the accident analysis.

22. Part ss. Revise the acceptance criteria for TVA-28 to ensure that technical specification tolerances can be verified.

23. Part vv. Your response in test description TVA-38 is not acceptable. It is the staff's position that the feedwater and feedwater heating system be tested to verify that main feedwater flow, pump head, and heater operation is within the parameters described in the FSAR and assumed in the accident analysis. Expand the test description to include this testing. Also identify any testing that will be conducted to verify that the main feedwater check valve water hammer (described in your report persuant to 10 CFR 50.55(e) dated May 21, 1979 has been corrected.

24. Part ww. It is the staff's position that valve closure time testing should be done at hot conditions. Modify test description TVA-40 to reflect this position. Also, expand the test description and acceptance criteria to include the time response of instrument lines, sensors, etc. to ensure that the ten seconds from initiation of an accident to valve closure assumed in the accident analysis can be met.

25. Part aaa. Modfify test descriptions TVA-21, CVCS, 31 Process Radiation Monitor, and 44A Liquid Waste Drains to provide acceptance criteria consistent with Reactor Coolant Pressure Boundary leak detection criteria.

26. Part o. In test description W9.1, describe the testing of control functions such as rod withdrawal inhibit features. Provide acceptance criteria for the tests.

27. Part r. It is the staff's position that rod deviation, rod insertion limits, and urgent failure alarms are necessary during operation to ensure core parameters remain within accident analysis assumptions. As such, they should be tested to ensure proper operation. Modify test description W5.4 to include this testing. Also, remove the reference to part-length rod banks.

28. Part t. Revise test description W8.2 to include acceptance criteria as requested.

29. Part z. Revise test abstract TVA-1 to include the acceptance criteria for response time and leak tightness as requested.

30. Part gg. It is the staff's position that all communications systems included in the station

emergency plan be tested to ensure they can be heard at necessary stations during operating conditions. Testing should include backup AC, DC, or self-contained battery power supplies. Remove the test description of HEPA filter testing in test TVA-11A and replace it with the appropriate test description.

31. Part hh. Revise test descriptions TVA-12B and 12C to indicate that emergency loads will be demonstrated operable from offsite power sources to ensure control and protective circuitry functions properly. Revise the acceptance criteria for TVA-12A through 12D to reference appropriate FSAR design criteria.

32. Part jj. Modify test descriptions TVA 13A through C and 14A through E, as necessary, to comply with Regulatory Guide 1.108, position C2.

33. Parts kk and ll. As detailed in Regulatory Guide 1.68, one purpose of the preoperational test program is to verify that systems have been designed and constructed properly. The tests to ensure redundancy and correct load assignment fulfill part of this purpose. It is the staff's position that redundancy and load group assignments should be verified by these tests. Additionally, testing of DC loads at minimum battery terminal voltage assures valid design and procurement specifications were used and that installation was correct. It is the staff's position that DC loads should be tested at minimum battery voltage. Modify test description TVA-16 to reflect this position. Also, modify the test acceptance criteria to ensure that individual cell limits are not exceeded during the discharge test.

<u>Response:</u>

1. Part a

See revised FSAR Table 14.2-1 description for test W-1.1. This revision is intended to clarify test requirements for the RCS prior to and during the initial system heatup to hot functional test (HFT) conditions. Note that 'operability' testing of RCS components (valves, breakers, instrument loops, alarms, etc.) will be performed prior to HFT and will be associated with control logic verification. The acceptance criteria for operability test will be that the component operates in accordance with the design logic. Note that the PORV's are tested in W-1.2.

Acceptance criteria for minimum and maximum flow rates for the steam generator atmospheric dump valves and the pressurizer PORV's is considered by TVA to be unnecessary. As described in Section 5.2.2.3, the PORV's are designed to perform a control (not safety) function and to limit certain design pressure transients to less than the high pressure reactor trip setpoint. However, operation of the PORV's is not assumed in RCS pressure transient analyses addressed in Chapter 15. The only feasible requirement for verification of a maximum PORV capacity would be to validate the basis for the accident analyses of Section 15.2.12, Accidental Depressurization of the Reactor Coolant System. TVA does not believe it is necessary to verify be testing that the maximum relief capacity of a 6' ASME rated pressurizer safety valve is greater than that of a 3' PORV.

2. Part b

See revised FSAR Table 14.2-1 description of test W-1.2. This revision is intended to clarify test requirements during HFT. Note that no acceptance criteria is given for the relief tank temperature nor for the 240-hour RCS full flow test. Test requirements for the PRT are given in Test W-1.4. The requirement for 240 hours of full flow operation is a Westinghouse recommended minimum duration for HFT (including W-1.1, W-1.2, and W-1.3) and is associated with run in time on the RCP's. No acceptance criteria is necessary.

3. Part c

See revised FSAR Table 14.2.1 test objectives for test W-1.3.

4. Part e

See revised FSAR Table 14.2-1 test objectives for test W-1.8.

5. Part h

See revised FSAR Table 14.2-1 test objectives for test W-2.2.

6. Part i

See revised FSAR Table 14.2-1 test objectives for test W-2.1.

7. Part j

Test objective (5) 'Verification of effective closure of the RCS to SIS isolation check valves' of W-3.1 will be demonstrated in accordance with our response to question 212.74 in that this procedure has been incorporated into test instruction W-3.1A3. The test objectives of this instruction have been modified to reflect this testing. In the test description for the cold leg accumulator instrumentation the term applicable TVA documentation is in reference to the acceptance criteria identified in the TVA scoping document for W-3.2 which was originally provided by Westinghouse under document WAT/WBT-SU-2.3.2 Rev. 1 or later.

NOTE: Because of numbering system discrepancies between the test instructions and scoping documents, scoping document W-3.2 will be changed to W-3.1 (Part B) and W-3.3 to W-3.1 (Parts C-E).

8. Part k

Test W-7.1 has been expanded into W-7.1A and W-7.1B. W-7.1A measures primary sensor response time. All primary sensors with response time requirements in the Standard Tech. Spec. (STS) are tested. The acceptance criteria is a segment of time from the total channel response time required by the STS. W-7.1B measures the response time of the RPS or ESF channel from the primary sensor input onward. The acceptance criteria is a segment of time from the total channel response time required by the STS (or FSAR). The times measured in 7.1A and 7.1B are added to times measured in other tests of equipment actuated by these channels and total RPS or ESF times are verified.

9. Part m

Response times are measured in W-7.1A and W-7.1B. W-7.3 verifies that the proper safeguards channel operates from a simulated set of input signals. The safeguards channel operation is verified by observing that the proper ESF actuation relay operates when each possible combination of input signals required by the logic is simulated. The acceptance criteria requires that each ESF relay and its associated annunciators, status lights, etc. are operated for all possible combinations of inputs required by the logic.

10. Part w

    See revised FSAR Table 14.2-1 test objectives for test W-9.8.

11. Part y

    See revised FSAR Table 14.2-1 test objectives for test W-10.9.

12. Part cc

    The inclusion of TVA 6 test is an oversight. There is no advantage gained in testing the Containment Air Return Fans simultaneous to hot functionals. The fans are only required to be capable of withstanding a high pressure-temperature atmosphere. It has been stated in amendment 30 that the vendor is required to conduct tests such that this feature is verified. In addition, at the time of hot functionals the ice condenser will be loaded, and operation of the air return fans would result in a melting of the ice, since hot air would be recirculated through the ice baskets. For this reason, the air return fans test must be conducted prior to ice loading to verify proper operation of the ice condenser doors. Therefore, it is not practical to comply with this position of the NRC request in question 4.13.12 12, Part cc.

    See revised test description for TVA-4, -5, and -7.

13. Part ee

    The Auxiliary Building Gas Treatment Systems and the Reactor Building Purge System, tested in TVA-9A and TVA-9B, respectively, performs no heating or cooling functions.

    See revised test description for TVA-9C.

14. Part ff

    See revised FSAR Table 14.2-1 test objective for tests TVA-9A, TVA-9B, and TVA-9C.

15. Part ii

See revised FSAR Table 14.2-1 test objectives for test TVA-13C. Transfers between redundant divisions are not performed at Watts Bar.

16. Part mm

TVA-17 has been omitted from the Preoperational Test Program for Watts Bar Nuclear Plant, resulting from the mutual understanding that the Condenser Circulating Water System serves no safety-related function.

17. Part nn

See revised FSAR Table 14.2-1 test objectives for test TVA-18 (test objective 6).

18. Part oo

See revised FSAR Table 14.2-1 test objectives for test TVA-20 (test objective 6).

19. Part pp

See revised FSAR Table 14.2-1 test objectives for test TVA-22.

20. Part qq

See revised FSAR Table 14.2-1 test objectives for test TVA-25.

21. Part rr

See revised FSAR Table 14.2-1 test objectives for test TVA-27.

22. Part ss

See revised FSAR Table 14.2-1 test objectives for test TVA-28.

23. Part vv

See revised FSAR Table 14.2-1 test objectives for test TVA-38.

24. Part ww

The time response of the instrumentation will be tested in W-7.1. A total response time including valve closure is also tallied in W-1.7.

25. Part aaa - TVA will provide a response at a later date.

26. Part o

See revised FSAR Table 14.2-1 test objectives for test W-9.1.

27. Part r - TVA will provide a response at a later date.

28. Part t

See revised FSAR Table 14.2-1 test objectives for test W-8.2.

31. Part hh

The test which indicates that emergency loads will be demonstrated operable from offsite power sources to ensure control and protective circuitry functions properly is not done in TVA-12, but in TVA-13.

The FSAR has been modified to reflect design requirements for TVA-12A through 12D.

32. Part jj

See revised FSAR Table 14.2-1 test objectives for TVA 13A through C and 14E. Regulatory Guide 1.108, position C2 does not apply to the ventilating and heating systems of the Diesel Generator Building. However, a clarification of the acceptance criteria, appearing in the FSAR has been made for TVA-14C.

33. Part kk & ll

See revised FSAR Table 14.2-1 test objectives for test TVA-16.

40.108 Question
(9.5.8)

Provide the results of an analysis that demonstrates that the function of your diesel engine air intake and exhaust system design will not be degraded to an extent which prevents developing full engine rated power or cause engine shutdown as a consequence of any meteorological or accident condition. Include in your discussion the potential for, and effect of, fire extinguishing (gaseous) medium recirculation of diesel combustion products, products or combustion from a plant fire or other gases that may intentionally or accidentally be released on site, on the performance of the diesel generator. (SRP 9.5.8, Part III, Item 3).

Response

The results of the analysis by TVA concerning the effects of an accident condition on the function of the diesel generator air intake and exhaust systems has been previously provided in the Watts Bar Nuclear Plant FSAR, Section 9.5.8.3 and the Watts Bar Nuclear Fire Protection Submittal dated September 1980, forwarded to the NRC by letter from L. M. Mills to A. Schwencer dated September 9, 1980. Refer to the TVA response to NRC Questions 19, 36, and 50 for discussion of the Co_2 fire extinguishing agent used in the diesel generator building.

The results of the analysis by TVA concerning the effects of the CO_2 fire extinguishing medium, any possible recirculation of diesel combustion products or products of combustion from either plant fires or other gases upon the diesel generator intake and exhaust systems has been previously provided in TVA submittals. The Watts Bar Nuclear Plant Fire Protection Submittal dated September 1980, forwarded by letter to the NRC from L. M. Mills to A. Schwencer dated September 9, 1980, discusses these subjects in response to NRC Question 19, 36, and 50.

31.25 Question

The statement made on Page 3.10-2 regarding the power range neutron detectors seems to imply that the detectors have been vibration tested but not seismically tested in the same manner as the remainder of the equipment in the Westinghouse scope of supply. The type testing has not been described for these detectors. Amend this portion of the FSAR in a manner that provides all the information on the safety portions of the NTS as required in Section 3.10.2 of the Standard Format.

Response

See revised response to Question 31.49.

31.49 <u>Question</u>
(3.10)
(7.2)

The response to Question 31.25 appears to be incomplete. Justify not qualifying the detectors for the 0.5 to 6 Hz. range.

<u>Response</u>

In response to this question a report entitled 'Seismic Testing of the Nuclear Instrumentation System Power Range Detector for the Watts Bar Plant,' was submitted to the NRC by letter dated September 21, 1981 from L. M. Mills to E. Adensam.

40.114 Question
(10.2)

Your response to request 040.48 is incomplete in that you only provided a partial response on the protection to the turbine overspeed control system equipment, electrical wiring and hydraulic lines from the effects of a high or moderate energy trip failure so that the turbine overspeed protection system will not be damaged to preclude its safety function. You state that in the event of a high energy line break which damages either the mechanical or electrical overspeed trips that the other overspeed trip will still function. You also stated that no protection from high and moderate energy line breaks was provided for the electrical wiring and hydraulic lines of the turbine overspeed protection system. Show that in the event of a high or moderate energy line break which damages the electrical wiring and/or hydraulic lines of the turbine overspeed protection system that the safety function of the systems will not be impaired (i.e., failsafe system).

Response

A review of the high and moderate energy lines in the vicinity of the normal electrohydraulic control (EHC) and the overspeed protection system was performed. In the event of a line break, at least one of the control or overspeed systems would remain in service and prevent the turbine speed from exceeding the design overspeed (120 percent of rated speed).

The normal EHC system for speed/load control, the train A overspeed protection controller (OPC) system, the train B OPC system, the train A electrical overspeed trip system (the train B trip system will also be initiated by the train A trip), and the mechanical overspeed trip mechanism, all provide speed control protection so as to maintain turbine speed at safe operating levels. Any one of these systems when fully operational will prevent the turbine speed from exceeding 120 percent of rated speed. Consequently only one of the above systems needs to operate to perform the turbine overspeed protection system's intended function. A brief description of the necessary components of each system is given below.

The normal EHC system for speed/load control consists primarily of the EHC cabinet; the operator's panel; the governor and stop valves; the governor and stop valves servo-actuators and LVDT's; normal speed, impulse chamber pressure, and load unbalance sensors; and the high pressure fluid control system. The two, independent (train A or B) OPC systems each consists primarily of the

EHC cabinet; the governor and stop valves; the high pressure fluid control system; the OPC speed sensor; and a trained (train A or B) solenoid-operated, high pressure fluid dump valve. The train A electrical overspeed trip system consists primarily of the EHC cabinet; the governor and stop valves; the high pressure fluid control and autostop oil systems; the OPC speed sensor, and the solenoid-operated autostop oil dump valve. The train B trip system which activates on a train A trip signal consists primarily of the same components as the train A electrical overspeed trip system except that an additional trip relay (from train A) replaces the OPC speed sensor as the trip activation device and the auto stop oil system is not part of the system. The mechanical overspeed system is strictly mechanical (and hydraulic) and consists primarily of the governor and stop valves, the high pressure fluid control and autostop oil systems, and the mechanical overspeed mechanism. (A more detailed explanation of the turbine speed control and overspeed protection systems, including the systems discussed above, and the redundancy provided within these systems is provided in our response to Q40.112.)

The EHC cabinet (electronic controls), operator's panel, and the additional trip relay (from train A) for activation of the train B trip system are located in the control building; and consequently, would not be affected in the event of a high or moderate energy line break. The remaining components described above are located in the turbine building; and could be affected by a high or moderate energy line break. The following paragraphs discuss the effects, if any, of a high or moderate energy line break on the above turbine building components and the turbine overspeed protection system. The normal and OPC speed sensors and the mechanical overspeed trip mechanism are totally enclosed (within a casing approximately one inch thick) and located in the turbine front standard above the main turbine floor. The governor and stop valves, the governor and stop valves servoactuators and LVDT's, some of the high pressure fluid control and autostop oil piping, the load unbalance sensor, the solenoid-operated high pressure fluid dump valves (inside 1/8-inch enclosure), and the solenoid-operated sutostop oil pump valve (inside a thick metal casing which is also inside the 1/8-inch enclosure) are also located on or above the main turbine floor. The impulse chamber pressure sensor is located approximately 25 feet below the high pressure turbine just above the next floor below. The remainder of the high pressure fluid control piping is also located below the main turbine floor.

Electrical wiring for normal controls and electrical overspeed systems is routed between the above electrical components above the below the turbine floor to the EHC cabinet and other controls in the control building through penetrations one and two floors below the main turbine floor. The normal EHC speed/load control system wiring is located adjacent to each steam chest for the governor and stop valves servo-actuators and LVDT's associated with that particular steam chest. This wiring drops vertically through the main turbine floor and is routed by separate cable trays to the control building. The normal and OPC speed sensors wiring is adjacent to the outer turbine building wall near the B and C low pressure turbines. The wiring for the normal and OPC speed sensors, and the load unbalance sensor also drops down through the turbine floor at separate locations and are also routed separately to the main control building using separate paths. The impulse pressure sensor wiring is near the next floor below the HP turbine. This wiring also drops down through the floor and is routed to the control building. The train A solenoid operated autostop oil dump valve is located adjacent to one side of the front standard, and the train A and B (OPC) and train B trip solenoid operated high pressure fluid dump valves are located adjacent to the other side of the front standard. The trained wiring which connects these valves with the controls in the control building is routed through trained conduits which are physically and electrically separated similar to other trained systems in the auxiliary building and safety-related area. The wiring in the vicinity of the above dump valves is protected by the best means available while maintaining the interface with the turbine's high pressure fluid control and autostop oil systems in which the above dump valves are located.

High and/or medium pressure hydraulic (high pressure fluid control and autostop oil systems) lines are connected to each governor and stop, valve's servo-actuators, to each of the above solenoid-operated dump valves, and to the front standard (and mechanical overspeed trip mechanism). Since each governor and stop valve is spring closed, it is necessary only to dump the high pressure fluid (or autostop oil), or to rupture a high pressure fluid or autostop oil supply line to dump the high pressure fluid from under the servoactuators which cause all of the steam valves to close (note that the reheat stop and intercept valves will also close, but for the sake of simplicity only the governor and stop valves are discussed in this response).

It is conceivable that a high or medium energy line break could render the normal EHC speed/load control system and some of the overspeed systems inoperable; but it is inconceivable due to the steam piping configuration, the location of the hydraulic lines, and the number of overspeed systems provided that a single pipe break could render the overspeed protection system inoperable. If the line break occurred in the vicinity of the high pressure turbine above the main turbine floor, the broken line would almost certainly rupture a hydraulic line (which would trip the turbine) whereas the failure of all overspeed and normal speed control systems would be highly unlikely. A line break below the main turbine floor would be less severe since the mechanical overspeed mechanism and the trained dump valves are shielded by the turbine flooring and would provide overspeed protection.

Consequently, the turbine overspeed protection system is inherently safe and the probability of overspeed beyond design conditions is highly unlikely.

40.110 Question
(9.5.8)

Show by analysis that a potential fire in the diesel generator building together with a single failure of the fire protection system will not degrade the quality of the diesel combustion air so that the remaining diesel will be able to provide full rated power.

Response

The Watts Bar Fire Hazard Analysis, derived by a detailed review of plant design and an evaluation of the effects of postulated fires, was provided as part of the Watts Bar Nuclear Plant Fire Protection Program Reevaluation, forwarded by letter to the NRC from J. E. Gilleland to R. S. Boyd dated April 18, 1977, and supplemented by Watts Bar Nuclear Plant Fire Protection Submittal dated September 1980, forwared by letter from L. M. Mills to A. Schwencer dated September 9, 1980. The analysis for the diesel generator building was provided in this report in the TVA responses which included Table 1.1 and Figure 1-9 to NRC Questions 1, 19, and 50.

40.122 Question
(10.4.4)

Provide the results of an analysis indicating that failure of the turbine by-pass system high energy line will not have an adverse effect or preclude operation of the turbine speed control system or any safety related components or systems located close to the turbine by-pass system. (SRP 10.4.4, Part III, Item 4).

Response

As stated in Q40.114, line breaks below the main turbine floor are not expected to have any effect on the turbine speed control system and turbine overspeed protection system. The steam dump piping is located two floors below the main turbine bay and at least 35 feet away from the closest turbine (normal control) wiring. There are no other safety related systems or components in the vicinity of the steam dump piping. Consequently, a steam dump line break will not have any affect on safe shutdown of the turbine or any safety related equipment.

121.13 Question

The materials surveillance program uses six specimen capsules, containing reactor vessel steel specimens of the limiting base material, weld metal, and weld heat-affected zone material. To demonstrate compliance with Appendix H, 10 CFR Part 50, provide a table as illustrated in Enclosure 1 that includes the following information for each surveillance specimen capsule of both unit Nos. 1 and 2:

(1) The actual surveillance materials in each capsule;

(2) The beltline material from which each surveillance material was obtained;

(3) The weld wire, heat of filler material, flux type, lot of flux, and plate material used in each weldment test specimen;

(4) The test specimen types, number, and orientation;

(5) The fabrication history of each of the materials in the capsule;

(6) The actual location of each capsule in the reactor vessel;

(7) The lead factor for each capsule calculated with respect to the vessel inner wall;

(8) The proposed time of capsule withdrawal (effective full power years).

Response

Detailed information regarding the reactor vessel material surveillance program is contained in Westinghouse topical reports WCAP-9298 for Watts Bar unit 1 and WCAP-9455 for Watts Bar unit 2.

121.14 Question

Provide details of the program for calibrating temperature instruments and Charpy V-notch machines. This information should be sufficient to demonstrate that the program is in compliance with Paragraph NB-2360 of the ASME Code as required by Paragraph III.B.3, Appendix G, 10 CFR Part 50.

Response

Temperature instruments used to control the test temperature of specimens were calibrated once in each three month interval. Charpy impact machines were calibrated once a year with a standard test specimen from the U.S. Army Materials and Mechanics Research Center.

121.15 Question

Provide information, including training and experience, to demonstrate that the qualification of individuals who performed fracture toughness tests are in compliance with the requirements of Paragraph III.B.4, Appendix G, 10 CFR Part 50.

Response

Although formal training programs were not in effect at all of the primary pressure boundary component vendors at the time of the manufacture of the Watts Bar components, the training of personnel engaged in testing was emphasized by all vendors. This training, plus the on-the-job experience in performing these routine tests, provides assurance that these tests were correctly performed.

121.17 Question

To demonstrate the integrity of the reactor coolant pump flywheels, supply the Charpy V-notch impact and tensile data for each flywheel, explicitly stating the material used for each flywheel.

Also, confirm that welding, including repair welding, was not performed on any finished flywheel. If welding were performed, identify the flywheel(s) and location of the welds.

Response

The reactor coolant pump flywheels are A533, Grade B, Class 1 material. The material exhibits 'no-break' performance at 20°F, and at least 50 foot pounds Charpy V-notch energy at 70°F. No welding was performed on the flywheels. Additional information is provided in Section 5.2.6.

212.97 Question
(6.3)
(212.86)

Provide further information about preoperational tests to demonstrate UHI performance. Provide UHI accumulator values such as line resistance and isolation setpoint assumed in safety analysis (with tolerance bands) and acceptance criteria (including instrument uncertainty) for the preop tests. Describe how tests demonstrate that no nitrogen entraining vortices would degrade UHI performance.

Response

The Westinghouse UHIS pre-operational test program has consisted of both a low pressure blowdown (90 psig initial pressure) and a high pressure blowdown (1240 psig initial pressure). The low pressure blowdown portion of the test provides direct measurement of the relative resistance of the UHIS injection piping which with a specific UHIS hydraulic isolation valve (HIV) closure time establishes the water accumulator level instrument setpoint. The high pressure blowdown portion of the test was specified in order to: 1) verify the closing performance of the four hydraulic isolation valves (HIV's), and 2) verify that the water accumulator level instrumentation setpoint, as determined by the low pressure blowdown test, results in the delivery of the proper volume of water to the reactor vessel upper head. This high pressure blowdown test was utilized in order to provide fluid flow conditions that are more severe than the flow conditions expected during UHIS operation following the worst postulated loss of coolant accident. Thus, direct observations of the HIV operation and volume delivery could be made at conservatively high flow rates.

The high pressure UHIS blowdown tests performed at Sequoyah unit 1 and other plants have indeed provided direct verification that the UHIS HIV's perform as expected. These tests have also verified the correlation between the piping resistance of the UHIS injection piping (determined by the low pressure UHIS blowdown test), the closing time of the UHIS HIV and the water accumulator level setpoint required to obtain the desired delivered water volume.

The complete UHIS test program including both the low and high pressure blowdowns have been performed on the Sequoyah unit 1 UHIS. The low pressure test has determined the piping resistance of the unit 1 UHIS injection piping, the high pressure test has verified HIV valve closure and resultant water volume delivered. Since the Watts Bar units are mechanically and hydraulically identical to the Sequoyah unit 1 UHIS, the same level setpoint will provide the same delivered volume provided the HIV closure time is set to be identical to unit 1.

To provide actual verification of this similarity between the units, Watts Bar will perform the low pressure blowdown portion of the UHIS pre-operational test program. The results of this test are to show that; 1) the resistance of the Watts Bar units 2 UHIS injection piping and 2) the HIV closing times are identical to Sequoyah unit 1's. Westinghouse has provided test acceptance criteria for these parameters, see Table Q212.97-1. By assuring that these criteria have been met and by utilizing the unit 1 water level setpoint, TVA will be assured that the volume delivered by the Watts Bar units, UHIS will be the same as Sequoyah unit 1 and will be within the delivered water volume band contained in the Sequoyah Appendix K ECCS analysis.

The benefits of not performing a high pressure blowdown test on the Watts Bar units UHIS would be two-fold:

1. The time required for UHIS testing would be shortended and thus would reduce the amount of test personnel time required and also would make the reactor coolant system available for other necessary pre-operational work and testing.

2. The need for valve disk and seats repair or replacement after the test would be eliminated. Although all the UHIS isolation valves have been modified to minimize valve damage during the high pressure blowdown test, it is recognized that the UHIS isolation valves are subjected to very severe ΔP and flow conditions during the test. These test conditions far exceed that which the valves would see during the worst possible LOCA recovery situation. Thus, although the 'post-test' valve leakage has been acceptable for system isolation following a LOCA, valve repair was required to reduce valve leakage for system isolation when the RCS was depressurized for refueling, etc.

In the safety analysis the UHI accumulator discharge line resistance is assumed to be a fl/D of 23.0 and the accumulator isolation valve closure setpoint is 945 $ft^3 \pm 50\ ft^3$ of water discharged into the reactor vessel.

The UHI water volume uncertainty band applied analysis is developed as a bound to the possible variation in the total UHI water delivered. Variation in total delivery is postulated to result from two causes: variation which results from measurement error and the uncertainty associated with system performance and variation which results from a single failure in the UHI system.

Each source of variation in total UHI delivery and its associated volume contribution is given in attached Table Q212.97-2. The volume contributions associated with single failure results from a postulated system malfunctions and are therefore not subject to statistical consideration. The single failure volumes are added directly to the nominal UHI volume setpoint. The remaining volume sources are the result of system uncertainties.

Nitrogen samples of the discharged fluid are taken just before and during the UHI hi-pressure blowdown. These samples have consistently demonstrated no significant amount of nitrogen is intrained in the discharged fluid.

TABLE Q212.97-1

WATTS BAR UNITS 1 AND 2 LOW PRESSURE BLOWDOWN
TEST ACCEPTANCE CRITERIA

a) Relative resistance - 3.28 ± .164

b) HIV static opening time 4.0 seconds ± 0.25

c) HIV static closing time - 3.5 seconds ± 0.05

d) Level setpoint for actuation of HIV - 107.5 inches ± .25 atmospheric pressure

TABLE Q212.97-2

UHI WATER VOLUME UNCERTAINTY

SOURCE	ASSOCIATED VOLUME FT^3
SINGLE FAILURE	
TRAIN FAILURE	+55
ONE VALVE CLOSES PREMATURELY	-51
TANK LEVEL INSTRUMENTATION ACCURACY	$\pm$ 2.2
INSTRUMENT SETTING TOLERANCE	$\pm$ 2.0
TANK VOLUME TOLERANCE	$\pm$ 19
HYDRAULIC ISOLATION VALVE STROKING TIME	$\pm$ 12.
TANK LEVEL READING ACCURACY	$\pm$ 11.
TOTAL ERROR	198.4
TOTAL ERROR AT .95 PROBABILITY	156.

31.112 <u>Question:</u>
(Add)
(3.11)
(15.3.1.1)
(Q040.2)
(T3.1103)

The staff is concerned that Class IE equipment inside of the containment which is not required for a particular accident or which has completed its protective function may not satisfy the requirements of IEEE Std 279-1971, Section 4.20. Therefore:

1. Describe how the operator can determine if a particular instrument, under these conditions, is giving a true or a false indication.

2. Describe how spurious actuation of undesired systems responses, which could result from these failures, are prevented.

3. Describe the consequences of a 17.5 lbm/sec leak in the pressurizer area or its connected lines on the instrumentation and controls for the pressurizer. (e.g., Class IE pressurizer level and pressure sensors, other sensors which are associated with the control of the pressurizer spray, heaters, let down flow, and power-operated relief.)

<u>Response:</u>

1., 2. The Electrical Equipment Qualification Report for NUREG-0588 will consider the problem presented in parts 1 and 2. NUREG-0588 requires that all Class IE electrical equipment inside containment must be evaluated for environmental qualification for the most severe environment they will experience.

In particular, the evaluation must (per App. E of NUREG-0588) consider, in part, equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand any accident environment for the time during which it must not fail with safety margin to failure. Thus, all instruments that may, if failed, misrepresent information needed to mitigate the accident will be qualified for the environment.

3. The effects of leaks on the pressurizer instrumentation and controls depend on the location of the leak. The areas that must be considered are (a) at the top of pressurizer enclosure, (b) in the immediate area of each panel containing pressurizer instrumentation, and (c) on the power supply for the pressurizer heaters. A 17.5 lbm/sec leak is the maximum break size for which the normal makeup system can maintain pressurizer level. A leak of this size would cause jet impingements and environmental influences only in a limited area near the break. The evaluation includes both Class IE and non-Class IE instrumentation and controls for the pressurizer.

A. <u>Leak at the Top of the Pressurizer Enclosure</u>

A leak in this area could cause the temperature in the enclosure to exceed the design temperature of mechanical components and instrumentation located at the top of the enclosure.

The instrumentation and components located at the top of the enclosure are:

1. Pressurizer upper level tops and pressure tops.

2.	PCV-68-334	Pressurizer Power Operated Relief Valve (PORV)
	PCV-68-340A	Pressurizer PORV
	FCV-68-332	Pressurizer PORV Block Valve
	FCV-68-333	Pressurizer PROV Block Valve
3.	XE-68-340A	Valve Postion Accoustic Monitor
	XE-68-334	Valve Postion Accoustic Monitor
	XE-68-363	Valve Postion Accoustic Monitor
	XE-68-364	Valve Postion Accoustic Monitor

XE-68-365	Valve Postion Accoustic Monitor
TE-68-319	Pressurizer Water Temperature
TE-68-324	Pressurizer Vapor Temperature
TE-68-328	Pressurizer Relief Line Temperature
TE-68-329	Pressurizer Relief Line Temperature
TE-68-330	Pressurizer Relief Line Temperature
TE-68-331	Pressurizer Relief Line Temperature

1. An increased temperature of the pressurizer upper level taps and pressure taps will have an insignificant effect on the level indication and no effect on the pressure indication. The instrument lines pass through the pressurizer enclosure wall at approximately the same elevation as the level taps and condensing pots. The instrument sensing lines pass out of the affected zone to instrument panels. The effect of heating up the water in the lines for a short vertical distance will cause an insignificant error. The pressure readings will not be affected by the heating of the instrument sensing lines.

2. The pressurizer PORV's and PORV block valves will be reviewed in the WBN 'Electrical Equipment Environmental Qualification Report' in accordance with NUREG-0588 to ensure they do not fail in a way which is adverse to safety during a LOCA environment inside containment. The report does not consider the environmental conditions within the direct jet of this type of small leak. The PORV is a 'fail-closed' valve and the block valves are 'fail as-is' motor operated valves. The most probable failure mode would be for these valves to

go to their failed position which would cause no affects on the pressurizer operation. The worst possible failure conditions would be for the PORV's and PORV block valves to fail to open. Although the severity of the event would be increased, the Emergency Core Cooling System (ECCS) can provide adequate core cooling for these types of pressurizer steam-space LOCA's.

3. The accoustic monitors and temperature elements have no control functions and only provides information to the operators.

B. <u>Leaks in the Immediate Area of Each Panel Containing Pressurizer Instrumentation</u>

The panels containing pressurizer level and pressure transmitters which perform control functions are L-62, L-63, L-64, and L-179. Three of these panels, L-62, L-63, and L-179, also contain the pressure transmitters required for the Safety Injection Actuation Signal (SIAS) for a LOCA. The Pipe Rupture Evaluation required by Regulatory Guide 1.46 will ensure these panels are not affected by any postulated failures. The logic for the SIAS is a 2 out of 3 logic. The requirement to assume a single failure for the pipe break evaluation requires that none of the transmitters can be damaged.

Panel L-64 only contains one pressure transmitter used for the pressurizer control. PT-68-322 on Panel L-64 provides an interlock signal to one pressurizer PORV and a low-pressure alarm. The failure of this transmitter will not cause or change any control actions for the pressurizer.

C. <u>Leak on the Power Supply for the Pressurizer Heaters</u>

If a leak directly impinged on pressurizer heater power cables, the insulation could break down causing an electrical short. The heater breaker would open to de-energize the heaters. If one of the heater breaker fails to open (single failure), then the breaker to the 6.9 kV shutdown board could open because of the short. The loss of an electrical board is within the design basis of the plant. In the unlikely event that all the heaters are damaged (they enter the pressurizer in a common area), the plant could still be shutdown to cold shutdown as discussed in the response to Question 212.93.

40.111 Question:
(9.5.8)

Experience at some operating plants has shown that diesel engines have failed to start due to accumulation of dust and other deleterious material on electrical equipment associated with starting of the diesel generators (e.g., auxiliary relay contacts, control switches, etc.). Describe the provisions that have been made in your Diesel Generator Building design, Electrical Starting System, and combustion air and ventilation air intake design(s) to preclude this condition to assure availability of the diesel generator on demand.

Also describe under normal plant operation what procedure(s) will be used to minimize accumulation of dust in the diesel generator room; specifically, address concrete dust control. In your response also consider the condition when unit 1 is in operation and unit 2 is under construction (abnormal generation of dust).

Response:

Combustion air and ventilation air intakes are approximately 20 feet above ground level. Also, the combustion air system includes an oil-bath type filter. These features should restrict introduction of dust into the Diesel Generator Building.

In addition, provisions are made in the control relaying and switches to protect these items from accumulation of dust or other deleterious material by means of dust covers or enclosures.

40.118 Question:
(10.4.1)

Discuss the measures taken for detecting, controlling, and correcting condenser cooling water leakage into the condensate stream. Provide the permissible cooling water in-leakage and time of operation with in-leakage to assure that condensate/feedwater quality can be maintained within safe limits.

Response:

Each condenser is equipped with a sampling system that continuously monitors the condensate cation conductivity. A given increase in cation conductivity at one or more of the nine sampling points may indicate condenser cooling water inleakage.

Since each unit's condenser waterbox is divided into two sections, one section can be isolated during unit operation. Each unit has the capability of operating at a reduced power level while one-half of its condenser waterboxes are isolated. Repairs and/or plugging of defective tubes can be accomplished as soon as leaks are detected.

Any impurities in the condenser cooling water which are introduced into the condensate stream by condenser inleakage are removed by the Condensate Demineralizer System (CDS). The CDS is capable of maintaining the condensate/feedwater quality within the specified limits during a continuous inleakage of up to 15 gpm (total inleakage into either or both unit condensers). When the condenser inleakage is greater than 15 gpm, the leak must be located as soon as possible, the affected condenser section isolated and the leak repaired. The time required to detect condenser inleakage and to isolate a condenser section for corrective action is 24 hours maximum. Each unit's CDS is capable of maintaining the condensate/feedwater quality for six hours with condenser inleakage of up to 180 gpm.

31.44 Question:
(25)

Clarify the discrepancy between the statement, in FSAR Section 6.2.1.1.1, that 'Inadvertent containment spray system initiation during normal operation is not considered credible' and the reset design which is presented in FSAR Section 7.3.2.2.6.

Response:

See revised Section 6.2.1.1.1.

22.60 Question:
(6.2.4)

Standard Review Plan 6.2.4, 'Containment Isolation System,' paragraph II.5 requires that all power-operated isolation valves have position indication in the main control room. Table 6.2.4-1 indicates that the following containment isolation valves do not have position indication. We will require that position indication in the main control room be provided for these valves prior to plant operation:

a. outside containment isolation (CI) valve in two control air I&C lines (primary containment penetration Nos. 34 M-A and 34 M-B),

b. outside CI valve in the reactor coolant drain tank and PRT to vent header line (primary containment penetration No. 45),

c. outside CI valve in the control air line (primary containment penetration No. 90),

d. both CI valves in the containment air monitor lines through primary containment penetration Nos. 94 M-A, B, and C and 94 M-A, B, and C,

e. outside remote manual CI valves in the upper head injection lines (primary containment penetration Nos.108 and 109), and

f. both CI valves in the upper head injection test line (primary containment penetration No. 110).

Response:

a. Penetration X-34 is not a multiple penetration at Watts Bar. A single control air line penetrates containment at this penetration. The outer isolation valve in this line does have position indication in the main control room. Table 6.2.4-1 has been updated to reflect penetration singularity and position indication of the outer isolation valve in the main control room.

b. Both the single inner and two outer isolation valves in the reactor coolant drain tank and PRT to vent header line (penetration X-45) have position indication in the main control room. Table 6.2.4-1 has been updated to reflect this data.

c. The outer isolation valve in the control air line (penetration X-90) does have position indication in the main control room. Table 6.2.4-1 has been updated to reflect this data.

d. All the inner and outer isolation valves on the containment air monitor lines through multiple penetrations X-94 and X-95 have position indication in the main control room. Table 6.2.4-1 has been updated to reflect this data.

e. Outside of containment, the upper head injection system (penetrations X-108, -109) is a closed system which employs a water seal. Thus, no containment isolation valves are employed outboard of containment in the main injection lines. A drain line and a lest line, however, attach to both injection lines outboard of containment. The lines are isolated from the closed system by containment isolation valves which have position indication in the main control room. Table 6.2.4-1 and Figure 6.2.4-22N have been updated to reflect this data.

f. All isolation valves in the upper head injection test line (penetration X-110) have position indication in the main control room. Table 6.2.4-1 has been updated to reflect this data.

121.11 Question:

Data supplied in FSAR Tables 5.2-11, 121.3-1, 121.7-1 through 121.7-4 and B 3/4.4-1 either do not meet or are not adequate enough to determine if the requirements of Appendix G, 10 CFR Part 50, are met. Therefore, to demonstrate compliance with Appendix G, supply the following information:

1. Identify, for both unit Nos. 1 and 2, each ferritic material, including base, weld, and weld heat-affected zone metal, used in a pressure-retaining component of the steam generators, pressurizer, and piping. Identify each material by specification and heat number and by location in the component;

2. Identify the unirradiated mechanical properties of each ferritic material requested in (1) as required by the testing programs of Section III of the ASME Code and Appendix G, 10 CFR Part 50. The test results should include Charpy V-notch, dropweight, and lateral expansion data.

3. Define an RT_{NDT} for each material identified in item (1) and clearly explain the method used to calculate each RT_{NDT} value.

For both unit Nos. 1 and 2 supply the following additional information for the ferritic materials in the reactor vessel beltline region, as defined by Paragraph II.H, Appendix G, 10 CFR Part 50:

4. Full Charpy V-notch curves, including data points, reported in impact energy and lateral expansion as a function of temperature;

5. The minimum upper shelf energy;

6. Chemical analyses, particularly Cu and P; and

7. Estimated change in RT_{NDT} and upper shelf energy as a function of neutron fluence.

From the data requested in Items 1 through 7, identify the most limiting material in each unit's reactor coolant pressure boundary at the beginning-of-life and at the end-of-life, and the procedure used to determine the limiting materials.

Response:

The Watts Bar unit 2 reactor vessel toughness data is provided in Table 121.11-1.

Complete Charpy V-notch data as a function of temperature is provided for the reactor vessel beltline forgings, weld, and heat affected zone material in Tables 121.11-2 and 121.11-3 for Watts Bar unit 1 and Tables 121.11-4 and 121.11-5 for Watts Bar unit 2.

Note that corrections have been made to the data shown for the intermediate and lower shells in Tables 5.2-11 and 121.7-2.

Since the reactor coolant pressure boundary vessel welds were made using SAW and SMAW process only, the conclusions of Westinghouse topical report WCAP-8291 regarding the properties of the heat affected zone materials do apply.

The limiting material is considered to be that material which will have the highest adjusted reference temperature at the end of life. The initial RT_{NDT} is obtained from the results of drop weight and Charpy tests. The materials are then evaluated on the basis of initial RT_{NDT}, neutron fluence at the end of life, and the predicted adjustment of RT_{NDT} based on chemical composition. The limiting material for each plant is the reactor vessel intermediate shell.

The ferritic pressure boundary materials of the pressurizer and steam generators were purchased to a maximum RT_{NDT} of 60°F. Drop weight and Charpy tests were performed to demonstrate that this maximum limit was not exceeded, although the actual RT_{NDT} was not necessarily identified. See the response to Question 121.12 for examples of actual test results obtained.

The root area of the intermediate to lower shell girth weld of Watts Bar unit 2 is a separate weld resulting from the grind out of the backing bar for the main weld. The root area weld is approximately 1/2 inch deep on the ID of the weld. The equipment specification required only three Charpy data points at 10°F. The results of these tests, along with data obtained from similar welds using similar procedures from the same vendor, provide assurance that the upper shelf for this root weld material exceeds 75 ft. lbs. and is not limiting for the vessel.

121.12 Question:

In Table 5.2-8 of the FSAR, it is indicated that SA 533 Class 2 steel has been used in the reactor coolant pressure boundary. Appendix G, 10 CFR Part 50, requires that the adequacy of the fracture toughness of materials with specified minimum yield strengths over 50 ksi be demonstrated to the Commission on an individual case basis. SA 533 Class 2 steel has a minimum specified yield strength of 70 ksi. Therefore, the applicant must identify each component and its parts that are made from SA 533 Class 2 steel. Furthermore, the applicant must address (a) the generic fracture toughness requirements for high strength materials as required by Section III of the ASME Code, and (b) the plant specific fracture toughness requirements of Appendix G, 10 CFR Part 50.

Response:

Materials with minimum yield strength over 50 ksi were used in the following applications in the Watts Bar reactor coolant pressure boundary.

Pressurizer - upper head
lower head
shell barrels
manway

Steam Generators - manway covers (for two generators on unit 1 and three generators on unit 2).

The conclusion of Westinghouse topical report WCAP-9292 are applicable to these materials. The specific fracture toughness data for these materials are shown in Table 121.12-1 for unit 1 and Table 121.12-2 for unit 2.

TABLE 121.12-1

WATTS BAR UNIT 1 - FRACTURE TOUGHNESS PROPERTIES OF SA 533 GN.A CL

Component	Part	2 NB Dropweight at °F	T_{NDT}	Charpy V-Notch Ft-Lb	Lateral Expansion Mils	Test Temp °F	RT_{NDT}
St. Gen. (1593)	Manway Cover	70	60	52, 58, 50	45, 46, 52	120	60
St. Gen. (1594)	Manway Cover	70	60	75, 79, 81	69, 68, 64	120	60
Pressurizer	Lower Head	20	10	64, 72, 64	66, 56, 56	70	10
	Upper Head	20	10	62, 78, 73	58, 63, 61	70	10
	Shell Barrel A	20	10	53, 61, 60	52, 52, 44	70	10
	Shell Barrel B	20	10	60, 70, 68	64, 54, 68	70	10
	Shell Barrel C	20	10	56, 52, 54	50, 51, 52	80	20
	Shell Barrel D	20	10	88, 76, 90	73, 75, 62	70	10
	Manway Cover	70	60	75, 79, 81	69, 68, 64	120	60

TABLE 121.12-2

WATTS BAR UNIT 2 - FRACTURE TOUGHNESS PROPERTIES OF SA 533 GN.A CL 2

Component	Part	2 NB Dropweight at °F	T_{NDT}	Charpy V-Notch Ft-Lb	Lateral Expansion Mils	Test Temp °F	RT_{NDT}
St. Gen. (1612)	Manway Cover	70	60	60, 63, 61	54, 54, 56	120	60
St. Gen. (1613)	Manway Cover	70	60	69, 74, 73	63, 63, 59	120	60
	Manway Cover	70	60	93, 91, 91	75, 74, 76	120	60
St. Gen. (1614)	Manway Cover	70	60	56, 51, 52	52, 51, 46	120	60
Pressurizer	Lower Head	20	10	70, 68, 76	66, 68, 68	70	10
	Upper Head	20	10	62, 66, 68	58, 60, 54	70	10
	Shell Barrel A	70	60	66, 77, 70	58, 68, 62	120	60
	Shell Barrel B	20	10	97, 85, 110	72, 78, 64	70	10
	Shell Barrel C	20	10	79, 88, 84	69, 74, 65	70	10
	Shell Barrel D	70	60	67, 68, 66	62, 64, 61	120	60
	Manway Cover	70	60	74, 78, 78	68, 72, 72	120	60

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TABLE Q121.11-5

WATTS BAR UNIT 2 REACTOR VESSEL BELTLINE REGION WELD
AND HEAT AFFECTED ZONE TOUGHNESS DATA
(Sheet 1)

WELD METAL				HAZ METAL			
Temp (°F)	Energy (ft lb)	Lat. Exp (Mils)	Shear (%)	Temp (°F)	Energy (ft lb)	Lat. Exp. (Mils)	Shear (%)
-184	1.5	4	0	-184	16.5	12	5
-166	5	4	0	-166	15	12	5
-148	6.5	8	5	-148	23	32	23
-130	10.5	12	16	-130	30.5	51	38
-112	19	20	17	-112	48	32	42
-94	8.5	12	10	-94	57.5	51	48
-76	13	16	25	-76	61	51	33
-58	22	24	30	-58	61	47	21
-40	25.5	28	30	-40	56.5	47	36
-22	42.5	39	40	-22	64	51	45
10	65	55	62	10	113	79	76
40	81	71	76	40	126.5	87	81
68	94	79	78	68	118.5	87	86
86	97.5	83	97	86	120	79	100
104	126	79	95	104	131.5	83	90
122	117.5	83	90	122	130	79	90
140	120.5	94	98	140	124.5	83	95
158	133	94	98	158	135	87	100

TABLE Q121.11-4

WATTS BAR UNIT 2 REACTOR VESSEL BELTLINE REGION SHELL FORGING TOUGHNESS DATA
(Sheet 1)

Intermediate Shell Forging 05
Heat No. 527828

Temp (°F)	Energy (ft lb)	Lat. Exp (Mils)	Shear (%)
-100	13	9	10
-100	7	3	6
-50	23	14	8
-50	20	12	8
10	43	30	20
0	58	44	27
0	32	25	20
25	74	56	39
40	49	37	30
40	66	52	42
74	94.5	67	70
74	67	52	45
74	80	59	55
125	95	72	80
125	109	80	100
210	120	84	100
210	100	76	100
210	110	74	100

Lower Shell Forging 04
Heat No. 528658

Temp (°F)	Energy (ft lb)	Lat. Exp. (Mils)	Shear (%)
-75	14	9	5
-75	17	11	5
-75	4	1	3
-30	33	24	20
-30	20	16	15
-10	55	40	23
-10	36	27	16
10	59.5	44	33
25	87	68	48
25	73	56	42
65	108	76	70
65	85	62	55
65	86	64	53
125	128	85	100
125	124	81	100
210	134	88	100
210	106	81	100
210	123	85	100

TABLE Q121.11-3

WATTS BAR UNIT 1 REACTOR VESSEL BELTLINE REGION WELD
AND HEAT AFFECTED ZONE TOUGHNESS DATA
(Sheet 1)

WELD METAL				HAZ METAL			
Temp (°F)	Energy (ft lb)	Lat. Exp (Mils)	Shear (%)	Temp (°F)	Energy (ft lb)	Lat. Exp. (Mils)	Shear (%)
-184	3	4	0	-184	19	20	4
-166	3.5	4	0	-166	23	20	5
-148	7.5	8	5	-148	19.5	12	5
-130	11	20	10	-130	32.5	24	17
-112	8	8	17	-112	52	35	23
-94	8.5	8	16	-94	37	28	21
-76	11.5	16	11	-76	52.5	39	36
-58	14.5	16	16	-58	65	47	42
-40	40.5	32	40	-40	82.5	55	58
-22	49	43	43	-22	95	71	78
10	77	63	69	10	110	79	81
40	91	71	75	40	115	75	76
68	109	87	85	68	129.5	83	86
86	118.5	87	95	86	125.5	87	100
104	130	79	90	104	129.5	94	98
122	113	79	88	122	127.5	90	90
140	118	87	92	140	121	95	95
158	129.5	91	98	158	128.5	92	92

TABLE Q121.11-2

WATTS BAR UNIT 1 REACTOR VESSEL BELTLINE REGION SHELL FORGING TOUGHNESS DATA
(Sheet 1)

Intermediate Shell Forging 05
Heat No. 527536

Temp (°F)	Energy (ft lb)	Lat. Exp (Mils)	Shear (%)
-100	5.5	2	2
-100	5	1	2
-100	6.5	3	2
-35	13	13	15
0	17	11	30
0	17	10	28
38	24.5	21	45
38	30	21	43
38	24	20	40
60	41	40	50
60	33	30	43
75	34	22	54
125	57	44	90
210	62	58	100
210	62	59	100
210	60	55	100
300	60	55	100
300	64	56	100

Lower Shell Forging 04
Heat No. 528522

Temp (°F)	Energy (ft lb)	Lat. Exp. (Mils)	Shear (%)
-50	36	20	12
-50	6	4	5
0	33	20	20
0	39	23	20
0	46	29	23
40	63	49	42
40	70	55	40
65	88	62	70
65	77	50	62
65	70	48	55
100	117	86	100
100	73	56	64
100	117	72	100
135	92	62	80
135	113	75	100
210	104	78	100
210	115	83	100
210	101	76	100

TABLE Q121.11-1

WATTS BAR UNIT 2 REACTOR VESSEL TOUGHNESS DATA
(Sheet 1)

Component	Heat No.	Mat'l Spec. No.	Cu %	P %	T_{NDT} °F	Minimum 50 Ft-lb 35 Mil Lat Exp Temp MWD °F	NMWD* °F	RT_{NDT} °F	Average Shelf Energy MWD Ft-Lb	NMWD Ft-Lb
Closure Head Dome	55994-1	A533B Cl.1	.08	.010	-4	-8	12	-4	>143	>99
Closure Head Ring	077793	A508 Cl.2	.08	.006	-4	28	48	-4	>138	>90
Closure Head Flange	528994	A508 Cl.2	.07	.009	-40	-13	7	-40	146	95
Vessel Flange	527944	A508 Cl.2	.06	.009	-22	10	30	-22	219	142
Inlet Nozzle	528209	A508 Cl.2	.05	.008	-22	5	25	-22	120	78
Inlet Nozzle	528267	A508 Cl.2	.06	.011	-22	26	46	-14	>101	>66
Inlet Nozzle	528267	A508 Cl.2	.06	.010	-13	23	43	-13	93	60
Inlet Nozzle	528329	A508 Cl.2	.04	.009	-13	25	45	-13	129	84
Outlet Nozzle	528095	A508 Cl.2	.06	.009	-22	17	37	-22	138	90
Outlet Nozzle	528207	A508 Cl.2	.06	.011	-13	7	27	-13	109	71
Outlet Nozzle	528209	A508 Cl.2	.05	.009	-40	5	25	-35	128	83
Outlet Nozzle	528329	A508 Cl.2	.04	.009	-31	26	46	-14	>128	>83
Nozzle Shell	411572	A508 Cl.2	.07	.005	-22	3	23	-22	142	92
Inter Shell	527828	A508 Cl.2	.05	.012	14	29	42**	14	137	110**
Lower Shell	528658	A508 Cl.2	.05	.006	5	14	20**	5	154	121**
Bottom Head Ring	528170	A508 Cl.2	.06	.009	-40	10	30	-30	160	104
Bottom Head Segment	55473-2	A533B Cl.1	.12	.006	-31	14	34	-26	111	72
Bottom Head Segment	55868-2	A533B Cl.1	.04	.010	-13	21	41	-13	152	99

TABLE Q121.11-1

WATTS BAR UNIT 2 REACTOR VESSEL TOUGHNESS DATA
(Sheet 2)

Component	Heat No.	Mat'l Spec. No.	Cu %	P %	T_{NDT} °F	Minimum 50 Ft-lb 35 Mil Lat Exp Temp MWD °F	NMWD* °F	RT_{NDT} °F	Average Shelf Energy MWD Ft-Lb	NMWD Ft-Lb
Bottom Head Segment	55888-2	A533B Cl.1	.04	.010	5	14	34	5	149	97
Bottom Head Dome	55979-2	A533B Cl.1	.04	.011	-13	14	34	-13	>120	>78
Inter to Lower Shell Girth Weld	55979-2	A533B Cl.1	.05	.010	-76	-	5	-55	-	126
Inter to Lower Shell Girth Weld (Root Area)	55979-2	A533B Cl.1	.03	.009	0*	-	-	0*	-	-

*Estimated per NRC Standard Review Plan MTB 5.3.2 'Pressure Temperature Limits'
**Based on Transverse Surveillance Program Data