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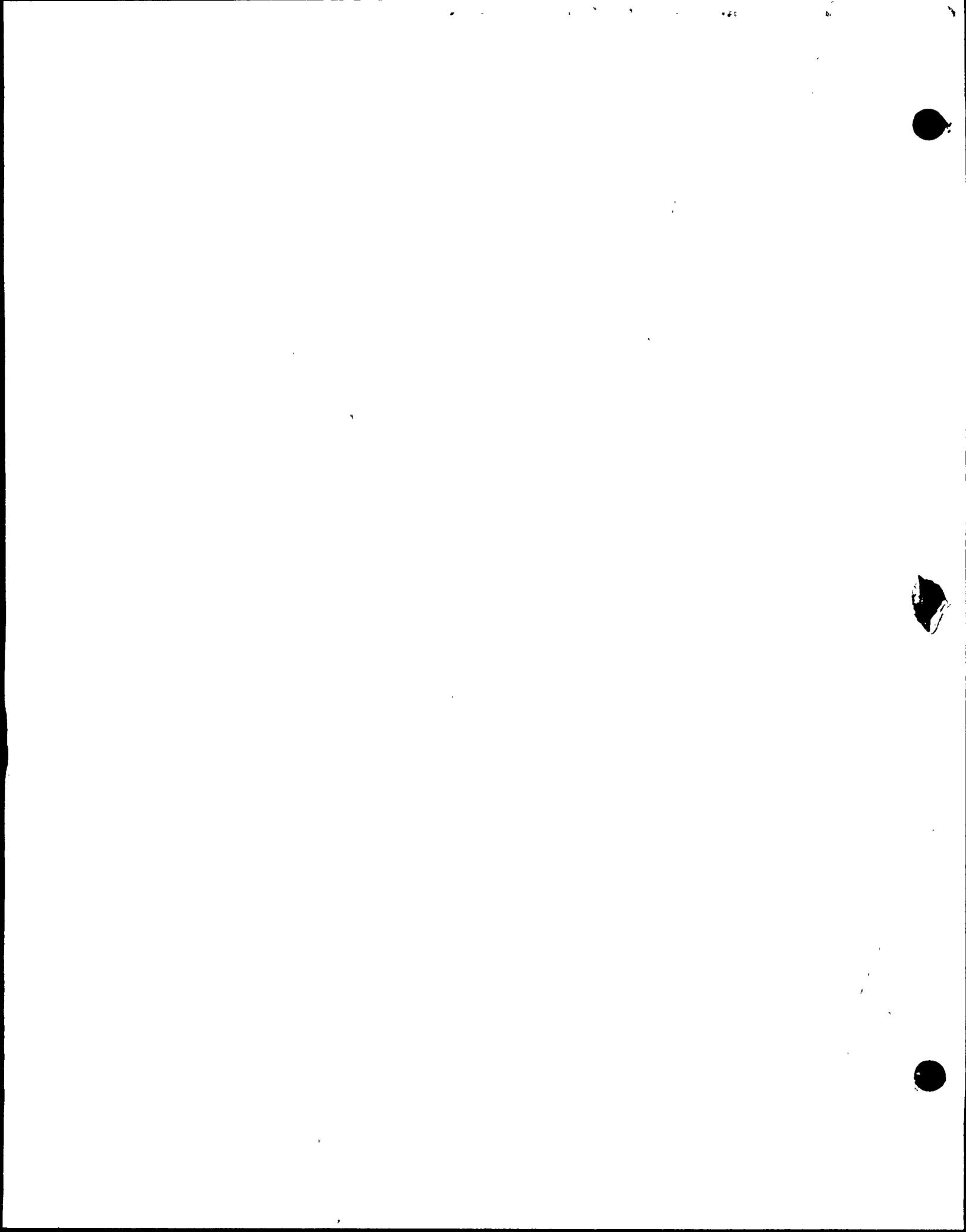
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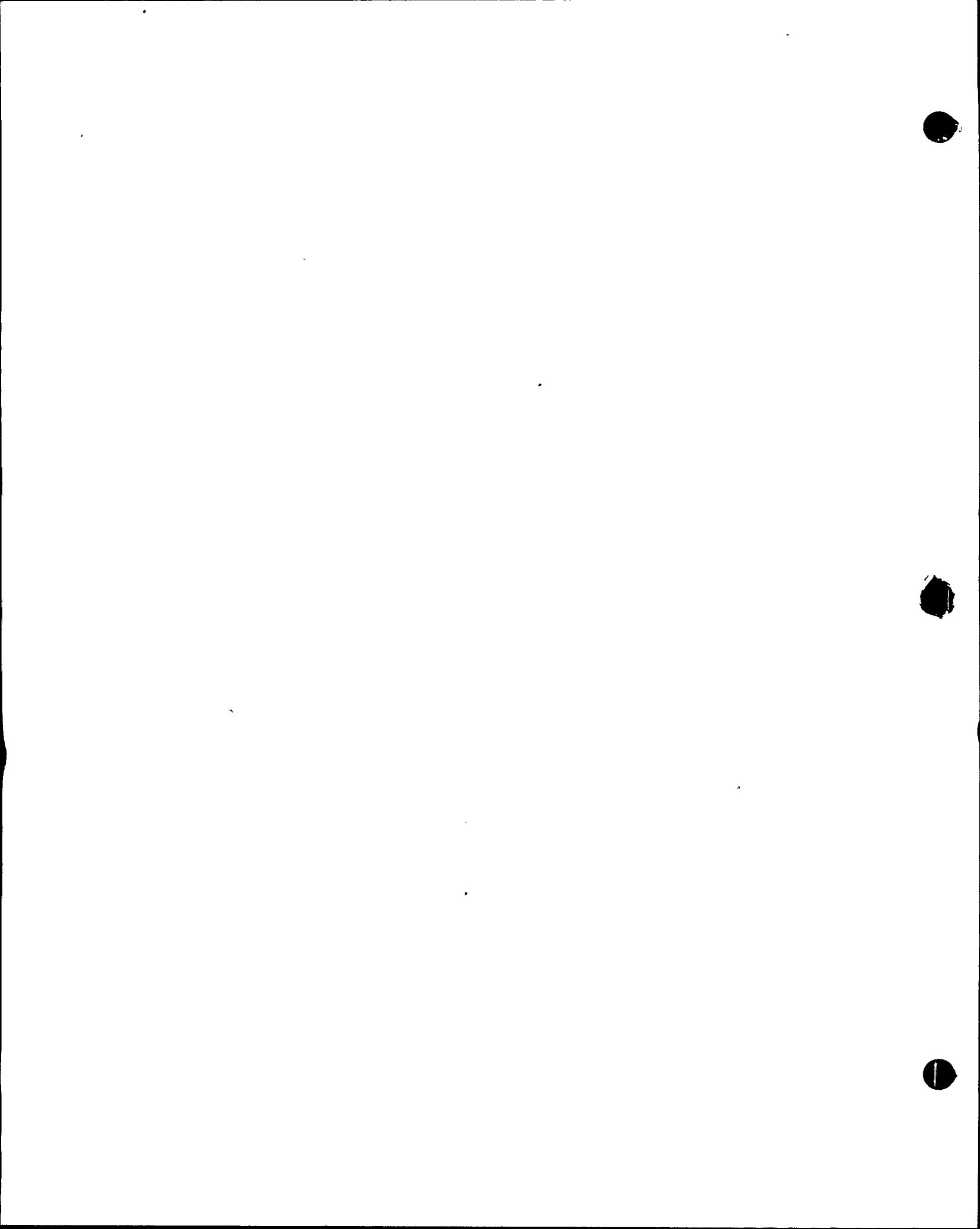
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BEN
Unit 1

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
1(9)	Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	≤ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate refueling floor c. Close atmosphere control system.
2(7) (8)	Instrument Channel SGTS Flow - Train A R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train B R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train C R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
1	Reactor Building Isolation Timer (refueling floor)	$0 \leq t \leq 2$ secs.	H or F	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
1	Reactor Building Isolation Timer (reactor zone)	$0 \leq t \leq 2$ secs.	G or A or H	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
2(10)	Group 1 (Initiating) Logic	N/A	A	1. Group 1: The valves in Group 1 are actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure

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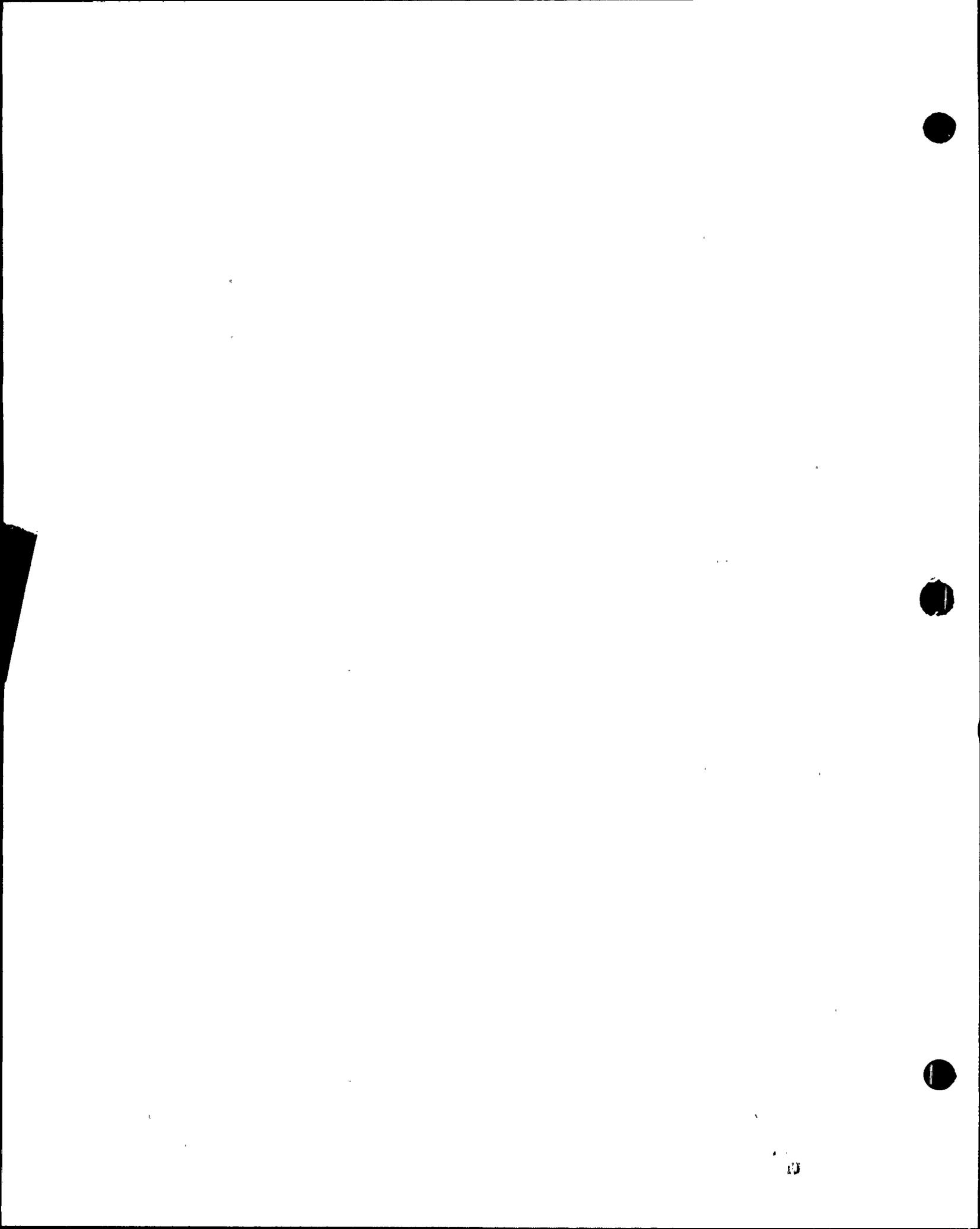


TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

BFN
Unit 1

Minimum No.
Instrument
Channels Operable
per Trip Sys(1)(11)

	Function	Trip Level Setting	Action (1)	Remarks
1	Group 1 (Actuation) Logic	N/A	B	1. Group 1: The valves in Group 1 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure
2	Group 2 (Initiating) Logic	N/A	A or (B and E)	1. Group 2: The valves in Group 2 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure
1	Group 2 (RHR Isolation-Actuation) Logic	N/A	D	
1	Group 8 (TIP-Actuation) Logic	N/A	J	
1	Group 2 (Drywell Sump Drains-Actuation) Logic	N/A	K	
1	Group 2 (Reactor Building & Refueling Floor, and Drywell Vent and Purge-Actuation) Logic	N/A	F and G	1. Part of Group 6 Logic
2	Group 3 (Initiating) Logic	N/A	C	1. Group 3: The valves in Group 3 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Reactor Water Cleanup System High Temperature c. Reactor Water Cleanup System High Drain Temperature

3.2/4.2-10



BFN
Unit 1

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
1	Group 3 (Actuation) Logic	N/A	C	
1	Group 6 Logic	N/A	F and G	1. Group 6: The valves in Group 6 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure c. Reactor Building Ventilation High Radiation
1	Group 8 (Initiating) Logic	N/A	J	1. Group 8: The valves in Group 8 are automatically actuated by only the following conditions: a. High Drywell Pressure b. Reactor Vessel Low Water Level 2. Same as Group 2 initiating logic.
1	Reactor Building Isolation (refueling floor) Logic	N/A	H or F	
1	Reactor Building Isolation (reactor zone) Logic	N/A	H or G or A	
1(7) (8)	SGTS Train A Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train B Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train C Logic	N/A	L or (A and F)	

Refer to Table 3.2.B for RCIC and HPCI functions including Groups 4, 5, and 7 valves.

3.2/4.2-11



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BFN
Unit 1

3.2/4.2-11a |



NOTES FOR TABLE 3.2.A (Cont'd)

4. Only required in RUN MODE (interlocked with Mode Switch).
5. Deleted
6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.
7. A train is considered a trip system.
8. Two out of three SGTS trains required. A failure of more than one will require actions A and F.
9. Deleted
10. Deleted
11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
12. A channel contains four sensors, all of which must be OPERABLE for the channel to be OPERABLE.

Power operations permitted for up to 30 days with 15 of the 16 temperature switches OPERABLE.

In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves.

13. The nominal setpoints for alarm and reactor trip (1.5 and 3.0 times background, respectively) are established based on the normal background at full power. The allowable setpoints for alarm and reactor trip are 1.2-1.8 and 2.4-3.6 times background, respectively.
14. Requires two independent channels from each physical location; there are two locations.

TABLE 3.2.B (Continued)

Unit 1	BFN	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
		2(2)	Instrument Channel - Reactor High Water Level	≤583" above vessel zero.	A	1. Above trip setting trips HPCI turbine.
		1	Instrument Channel - HPCI Turbine Steam Line High Flow	≤90 psi (7)	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
		4(4)	Instrument Channel - HPCI Steam Line Space High Temperature	≤200°F.	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
		3(2)	Instrument Channel - HPCI Steam Supply Pressure - Low (PS 73-1A-D)	≥100 psig	A	1. Below trip setting isolates HPCI system and trips HPCI turbine.
		3(2)	Instrument Channel - HPCI Turbine Exhaust Diaphragm (PS 73-20A-D)	≤20 psig	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
		1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
		1	RCIC System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. 2. Group 7: The valves in Group 7 are automatically actuated by only the following condition: 1. The respective turbine steam supply valve not fully closed.
		1	RCIC System (Isolation) Logic	N/A	B	1. Includes Group 5 valves. 2. Group 5: The valves in Group 5 are actuated by any of the following conditions: a. RCIC Steamline Space High Temperature b. RCIC Steamline High Flow c. RCIC Steamline Low Pressure d. RCIC Turbine Exhaust Diaphragm High Pressure
		1 (16)	ADS Logic	N/A	A	

3.2/4.2-19

TABLE 3.2.B (Continued)

Unit	BFN	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
3.2/4.2-20	Unit 1	1	RHR (LPCI) System (Initiation)	N/A	B	
		1	RHR (LPCI) System (Containment Cooling Spray) Logic	N/A	A	
		1	HPCI System (Initiating) Logic	N/A	B	<ol style="list-style-type: none"> 1. Includes Group 7 valves. 2. Group 7: The valves in Group 7 are automatically actuated by only the following condition: <ol style="list-style-type: none"> 1. The respective turbine steam supply valve not fully closed.
		1	HPCI System (Isolation) Logic	N/A	B	<ol style="list-style-type: none"> 1. Includes Group 4 valves. 2. Group 4: The valves in Group 4 are actuated by any of the following conditions: <ol style="list-style-type: none"> a. HPCI Steamline Space High Temperature b. HPCI Steamline High Flow c. HPCI Steamline Low Pressure d. HPCI Turbine Exhaust Diaphragm High Pressure
		1	Core Spray System auto initiation inhibit (Core Spray auto initiation).	N/A	B	<ol style="list-style-type: none"> 1. Inhibit due to the core spray system of another unit. 2. The inhibit is considered the contact in the auto initiating logic only; i.e., the permissive function of the inhibit.
		1	LPCI System auto initiation inhibit (LPCI auto initiation)	N/A	B	<ol style="list-style-type: none"> 1. Inhibit due to the LPCI System of another unit. 2. The inhibit is considered the contact in the auto initiating logic only, i.e., the permissive function of the inhibit.



3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at 378 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). These trip settings are adequate to prevent core uncover in the case of a break in the largest line assuming the maximum closing time. †

4.7.A. Primary Containment

4.7.A.2.g (Cont'd)

The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.C. Secondary Containment

3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

D. Primary Containment Isolation Valves

1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic



LIMITING CONDITIONS FOR OPERATION

3.7.D. Primary Containment Isolation Valves

2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:
 - a. The inoperable valve is restored to OPERABLE status, or
 - b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.
3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.D. Primary Containment Isolation Valves

4.7.D.1.a (Cont'd)

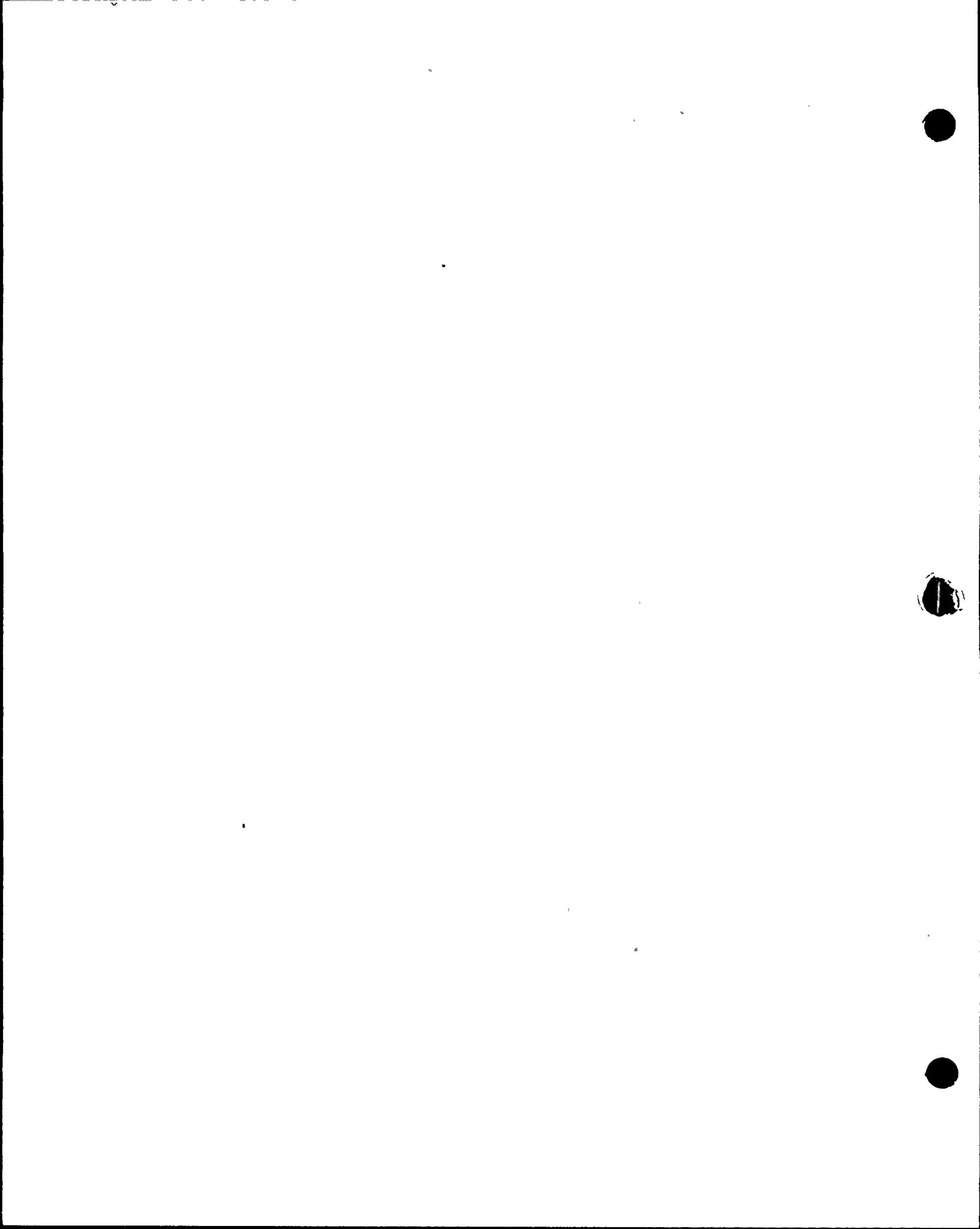
initiation and in accordance with Specification 1.0.MM, tested for closure times.

- b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested.

c. (Deleted)

- d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.



3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occurring.

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49.6 psig, P_a . As an added conservatism, the measured overall integrated leakage rate is further limited to $0.75 L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of three feet seven inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately three feet and water volume of approximately 123,000 cubic feet.



3.7/4.7 BASES (Cont'd)

Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached.

The maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and loss-of-coolant accident (LOCA) cases. The actions required by Specification 3.7.c., -d. -e. and -f., assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber ensures adequate margin for controlled blowdown anytime during RCIC operation and ensures margin for complete condensation of steam from the design basis LOCA.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a LOCA were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.



3.7/4.7 BASES (Cont'd)

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.1 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.06 feet to 3.58 feet will assure the integrity of the suppression chamber when subjected to post-loss-of-coolant suppression pool hydrodynamic forces.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a-percent or so) reaction of the zirconium and steam during a LOCA could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The <4 percent hydrogen concentration minimizes the possibility of hydrogen combustion following a LOCA.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the LOCA upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

To ensure that the hydrogen concentration is maintained less than 4 percent following an accident, liquid nitrogen is maintained onsite for containment atmosphere dilution. About 2,260 gallons would be sufficient as a seven-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2,500 gallons is conservative. Following a LOCA the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation. Failure of one system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

3.7/4.7 BASES (Cont'd)

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post-LOCA conditions. Post-LOCA calculations show that the CAD system initiated within two-hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.6-percent (at 4 hours) and 3.8-percent (at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism (\leq 3.8-percent), as a guide for CAD/Purge operations.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100-percent vacuum relief breakers (two parallel sets of two valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One reactor building vacuum breaker may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle the position indicating lights in the control room are designed to function as specified below:

Initial and Final Condition	Check - On	(Fully Closed)
	Green - On	
	Red - Off	
Opening Cycle	Check - Off	(Cracked Open)
	Green - Off	(> 80° Open)
	Red - On	(> 3° Open)
Closing Cycle	Check - On	(Fully Closed)
	Green - On	(< 80° Open)
	Red - Off	(< 3° Open)

The valve position indicating lights consist of one check light on the check light panel which confirms full closure, one green light next to the hand switch which confirms 80° of full opening and one red light next to the hand switch which confirms "near closure" (within 3° of full closure). Each light is on a separate switch. If the check light circuit is operable when the valve is exercised by its air operator there exists a confirmation that the valve will fully close. If the red light circuit is operable, there exists a

3.7/4.7 BASES (Cont'd)

confirmation that the valve will at least "nearly close" (within 3° of full closure). The green light circuit confirms the valve will fully open. If none of the lights change indication during the cycle, the air operator must be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully operable. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered operable the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten operable to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure and held constant. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.



3.7/4.7. BASES (Cont'd)

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a LOCA. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.



3.7/4.7 BASES (Cont'd)

Establishing the test limit of 2.0-percent/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate by 0.75 thereby providing a 25-percent margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 10 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, determining the oxygen concentration twice a week serves as an added assurance that the oxygen concentration will not exceed 4 percent.

3.7.B/3.7.C Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment, if required, when the reactor is shutdown and the drywell is open. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. All three standby gas treatment system fans are designed to

3.7/4.7 BASES (Cont'd)

automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage.

High efficiency particulate air (HEPA) filters are installed before and after the charcoal absorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine absorbers. The charcoal absorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal absorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal absorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal absorbers.

Only two of the three standby gas treatment systems are needed to clean up the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If more than one train is inoperable, all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel must be suspended and all reactors placed in a cold shutdown condition, because the remaining train would provide only 50 percent of the capacity required to filter and exhaust the reactor building atmosphere to the stack. Suspension of these activities shall not preclude movement of a component to a safe, conservative position. Operations that have the potential for draining the reactor vessel must be suspended as soon as practical to minimize the probability of a vessel draindown and subsequent potential for fission product release. Draindown of a reactor vessel containing no fuel does not present the possibility for fuel damage or significant fission product release and therefore is not a nuclear safety concern.

4.7.B/4.7.C Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip logic demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary

3.7/4.7 BASES (Cont'd)

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

3.7/4.7 BASES (Cont'd)

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the operable systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contain the list of all the Primary Containment Isolation Valves for which the TS requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. Section 1.0, Definitions, paragraph 0.1 allows normally locked and sealed-closed Primary Containment Isolation Valves to be opened to perform necessary operational activities under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated when a valid primary containment isolation signal is indicated.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level (378") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at 378" or main steam line high radiation.

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break



3.7/4.7 BASES (Cont'd)

in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - Process lines are closed only on the respective turbine steam supply valve not fully closed. This assures that the valves are not open when HPCI or RCIC action is required.

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

3.7/4.7 BASES (Cont'd)

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip logic are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability in accordance with Specification 1.0.MM results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested per Specification 1.0.MM to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25-inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage. During cycle 6, CREVS has been declared inoperable only because it does not meet its design basis for essentially zero unfiltered inleakage. Reactor power operations and fuel movement are acceptable until just prior to startup for unit 2 cycle 7. During cycle 6, CREVS must be demonstrated to be functional by performing all applicable surveillances. In the event that the applicable surveillances are not successfully performed, the actions required by the LCOs must be complied with.

High efficiency particulate absolute (HEPA) filters are installed prior to the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a

3.7/4.7 BASES (Cont'd)

limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to Cold Shutdown within 24 hours or refueling operations are terminated.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with USAEC Report-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

3.7.F/4.7.F Primary Containment Purge System

The Primary Containment Purge System is a non-safety related system that is normally isolated and normally not required to be functional during power operation. The system is designed to provide the preferred exhaust path for purging the primary containment system; however, the Standby Gas Treatment System can be used to perform the equivalent function.



3.7/4.7 BASES (Cont'd)

When the Primary Containment Purge System is in operation, the exhaust from the primary containment is first processed by a filter train assembly and then channeled through the reactor building roof exhaust system.

The filter train assembly contains a HEPA (high efficiency particulate air) filter, charcoal adsorber, and centrifugal fan. In-place tests are performed to ensure leak tightness of the filter train assembly of at least 99% and a HEPA efficiency of at least 99% removal of DOP particulates. Laboratory tests are performed on adsorber carbon samples to ensure an 85% removal efficiency for radioactive methyl iodide. Tests are performed to ensure that the system is not operating at a flow significantly different from the design flow, which may affect the removal efficiency of the HEPA filters and charcoal adsorbers. The pressure drop across the combined HEPA filters and charcoal adsorbers is checked once per operating cycle to be less than 8.5 inches of water at the system design flow rate to ensure that the filters and adsorbers are not clogged with excessive amounts of foreign matter.

The above tests are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with USAEC Report-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

If significant painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

The primary containment isolation valves associated with the purging of the primary containment are FCV 64-17, 64-18, 64-19, 64-29, 64-30, 64-32, 64-33, and 76-24. These valves may be open for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. Closure of these large diameter valves within the 24-hour period is needed to

3.7/4.7 BASES (Cont'd)

retain the reduced oxygen concentration specified in Technical Specification 3.7.A.5.b, and to minimize the time period which the primary containment is not isolated per the guidelines of Branch Technical Position CSB 6-4.

When the large diameter valves noted above are closed, primary containment venting is performed using valves FCV 64-31, 64-34, and 84-20 and the Standby Gas Treatment System. The operability of these primary containment isolation valves is governed by Technical Specification 3.7.D. The operability of the Standby Gas Treatment System is governed by Technical Specification 3.7.B.

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Unit 1

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UNIT 2
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TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

BFN
Unit 2

Minimum No.
Instrument
Channels Operable
per Trip Sys(1)(11)

	Function	Trip Level Setting	Action (1)	Remarks
1(9)	Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	≤ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate refueling floor c. Close atmosphere control system.
2(7) (8)	Instrument Channel SGTS Flow - Train A R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train B R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train C R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
1	Reactor Building Isolation Timer (refueling floor)	$0 \leq t \leq 2$ secs.	H or F	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
1	Reactor Building Isolation Timer (reactor zone)	$0 \leq t \leq 2$ secs.	G or A or H	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
2(10)	Group 1 (Initiating) Logic	N/A	A	1. Group 1: The valves in Group 1 are actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure

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TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Unit	BFN	Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	3.2/4.2-10	1	Group 1 (Actuation) Logic	N/A	B	1. Group 1: The valves in Group 1 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure
		2	Group 2 (Initiating) Logic	N/A	A or (B and E)	1. Group 2: The valves in Group 2 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure
		1	Group 2 (RHR Isolation-Actuation) Logic	N/A	D	
		1	Group 8 (TIP-Actuation) Logic	N/A	J	
		1	Group 2 (Drywell Sump Drains-Actuation) Logic	N/A	K	
		1	Group 2 (Reactor Building & Refueling Floor, and Drywell Vent and Purge-Actuation) Logic	N/A	F and G	1. Part of Group 6 Logic
		2	Group 3 (Initiating) Logic	N/A	C	1. Group 3: The valves in Group 3 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Reactor Water Cleanup (RWCU) System High Temperature in the main steam valve vault c. RWCU System High Temperature in the RWCU pump room 2A d. RWCU System High Temperature in the RWCU pump room 2B e. RWCU System High Temperature in the RWCU heat exchanger room f. RWCU System High Temperature in the space near the pipe trench containing RWCU piping

BFN
Unit 2

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
1	Group 3 (Actuation) Logic	N/A	C	
1	Group 6 Logic	N/A	F and G	1. Group 6: The valves in Group 6 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure c. Reactor Building Ventilation High Radiation
1	Group 8 (Initiating) Logic	N/A	J	1. Group 8: The valves in Group 8 are automatically actuated by only the following conditions: a. High Drywell Pressure b. Reactor Vessel Low Water Level 2. Same as Group 2 initiating logic.
1	Reactor Building Isolation (refueling floor) Logic	N/A	H or F	
1	Reactor Building Isolation (reactor zone) Logic	N/A	H or G or A	
1(7) (8)	SGTS Train A Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train B Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train C Logic	N/A	L or (A and F)	

Refer to Table 3.2.B for RCIC and HPCI functions including Groups 4, 5, and 7 valves.

3.2/4.2-11



NOTES FOR TABLE 3.2.A (Cont'd)

4. Only required in RUN MODE (interlocked with Mode Switch).
5. Deleted
6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.
7. A train is considered a trip system.
8. Two out of three SGTS trains required. A failure of more than one will require actions A and F.
9. Deleted
10. Deleted
11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
12. A channel contains four sensors, all of which must be OPERABLE for the channel to be OPERABLE.

Power operations permitted for up to 30 days with 15 of the 16 temperature switches OPERABLE.

In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves.

13. The nominal setpoints for alarm and reactor trip (1.5 and 3.0 times background, respectively) are established based on the normal background at full power. The allowable setpoints for alarm and reactor trip are 1.2-1.8 and 2.4-3.6 times background, respectively.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2(2)	Instrument Channel - Reactor High Water Level (LIS-3-208B and LIS-3-208D)	$\leq 583''$ above vessel zero.	A	1. Above trip setting trips HPCI turbine.
1	Instrument Channel - HPCI Turbine Steam Line High Flow (PDIS-73-1A and 1B)	≤ 90 psi (7)	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
3(2)	Instrument Channel - HPCI Steam Supply Pressure - Low (PS 73-1A-D)	≥ 100 psig	A	1. Below trip setting isolates HPCI system and trips HPCI turbine.
3(2)	Instrument Channel - HPCI Turbine Exhaust Diaphragm (PS 73-20A-D)	≤ 20 psig	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
1	RCIC System (Initiating) Logic	N/A	B	
1	RCIC System (Isolation) Logic	N/A	B	1. Includes Group 5 valves. 2. Group 5: The valves in Group 5 are actuated by any of the following conditions: a. RCIC Steamline Space High Temperature b. RCIC Steamline High Flow c. RCIC Steamline Low Pressure d. RCIC Turbine Exhaust Diaphragm High Pressure
1 (16)	ADS Logic	N/A	A	
1	RHR (LPCI) System (Initiation)	N/A	B	

BFN
Unit 2

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TABLE 3.2.B (Continued)

BFN Unit 2	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
	1	RHR (LPCI) System (Containment Cooling Spray) Logic	N/A	A	
	1	HPCI System (Initiating) Logic	N/A	B	
	1	HPCI System (Isolation) Logic	N/A	B	<ol style="list-style-type: none"> 1. Includes Group 4 valves. 2. Group 4: The valves in Group 4 are actuated by any of the following conditions: <ol style="list-style-type: none"> a. HPCI Steamline Space High Temperature b. HPCI Steamline High Flow c. HPCI Steamline Low Pressure d. HPCI Turbine Exhaust Diaphragm High Pressure
	1	Core Spray System auto initiation inhibit (Core Spray auto initiation).	N/A	B	<ol style="list-style-type: none"> 1. Inhibit due to the core spray system of another unit. 2. The inhibit is considered the contact in the auto initiating logic only; i.e., the permissive function of the inhibit.
	1	LPCI System auto initiation inhibit (LPCI auto initiation)	N/A	B	<ol style="list-style-type: none"> 1. Inhibit due to the LPCI System of another unit. 2. The inhibit is considered the contact in the auto initiating logic only, i.e., the permissive function of the inhibit.

3.2/4.2-20

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems.

The low water level instrumentation set to trip at ≥ 398 inches above vessel zero (Table 3.2.A) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). These trip settings are adequate to prevent core uncovering in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is ≥ 398 inches above vessel zero (Table 3.2.B)

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.C. Secondary Containment

3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

D. Primary Containment Isolation Valves

1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and in



3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.7.D. Primary Containment Isolation Valves

2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:
 - a. The inoperable valve is restored to OPERABLE status, or
 - b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.
3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.D. Primary Containment Isolation Valves

4.7.D.1.a (Cont'd)

- accordance with Specification 1.0.MM, tested for closure times.
- b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested.
 - c. (Deleted)
 - d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.
2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.



3.7/4.7 BASES

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occurring.

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49.6 psig, P_a . As an added conservatism, the measured overall integrated leakage rate is further limited to $0.75 L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of three feet seven inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately three feet and a water volume of approximately 123,000 cubic feet.

3.7/4.7 BASES (Cont'd)

Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached.

The maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and loss-of-coolant accident (LOCA) cases. The actions required by Specifications 3.7.C. - 3.7.F. assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber ensures adequate margin for controlled blowdown anytime during RCIC operation and ensures margin for complete condensation of steam from the design basis LOCA.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a LOCA were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.



3.7/4.7 BASES (Cont'd)

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.1 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.06 feet to 3.58 feet will assure the integrity of the suppression chamber when subjected to post-loss-of-coolant suppression pool hydrodynamic forces.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a-percent or so) reaction of the zirconium and steam during a LOCA could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The <4 percent hydrogen concentration minimizes the possibility of hydrogen combustion following a LOCA.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the LOCA upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

To ensure that the hydrogen concentration is maintained less than 4 percent following an accident, liquid nitrogen is maintained onsite for containment atmosphere dilution. About 2,260 gallons would be sufficient as a seven-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2,500 gallons is conservative. Following a LOCA the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation. Failure of one system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

3.7/4.7 BASES (Cont'd)

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post-LOCA conditions. Post-LOCA calculations show that the CAD system initiated within two-hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.6-percent (at 4 hours) and 3.8-percent (at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism (\leq 3.8-percent), as a guide for CAD/Purge operations.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100-percent vacuum relief breakers (two parallel sets of two valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One reactor building vacuum breaker may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle the position indicating lights in the control room are designed to function as specified below:

Initial and Final Condition	Check - On	(Fully Closed)
	Green - On	
	Red - Off	
Opening Cycle	Check - Off	(Cracked Open)
	Green - Off	(> 80° Open)
	Red - On	(> 3° Open)
Closing Cycle	Check - On	(Fully Closed)
	Green - On	(< 80° Open)
	Red - Off	(< 3° Open)

The valve position indicating lights consist of one check light on the check light panel which confirms full closure, one green light next to the hand switch which confirms 80° of full opening and one red light next to the hand switch which confirms "near closure" (within 3° of full closure). Each light is on a separate switch. If the check light circuit is operable when the valve is exercised by its air operator there exists a confirmation that the valve will fully close. If the red light circuit is operable, there exists a



3.7/4.7 BASES (Cont'd)

confirmation that the valve will at least "nearly close" (within 3° of full closure). The green light circuit confirms the valve will fully open. If none of the lights change indication during the cycle, the air operator must be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully operable. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered operable the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten operable to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure and held constant. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

3.7/4.7 BASES (Cont'd)

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a LOCA. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

3.7/4.7 BASES (Cont'd)

Establishing the test limit of 2.0-percent/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate by 0.75 thereby providing a 25-percent margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 10 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, determining the oxygen concentration twice a week serves as an added assurance that the oxygen concentration will not exceed 4 percent.

3.7.B/3.7.C Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment, if required, when the reactor is shutdown and the drywell is open. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. All three standby gas treatment system fans are designed to



3.7/4.7 BASES (Cont'd)

automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage.

High efficiency particulate air (HEPA) filters are installed before and after the charcoal absorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine absorbers. The charcoal absorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal absorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal absorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal absorbers.

Only two of the three standby gas treatment systems are needed to clean up the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If more than one train is inoperable, all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel must be suspended and all reactors placed in a cold shutdown condition, because the remaining train would provide only 50 percent of the capacity required to filter and exhaust the reactor building atmosphere to the stack. Suspension of these activities shall not preclude movement of a component to a safe, conservative position. Operations that have the potential for draining the reactor vessel must be suspended as soon as practical to minimize the probability of a vessel draindown and subsequent potential for fission product release. Draindown of a reactor vessel containing no fuel does not present the possibility for fuel damage or significant fission product release and therefore is not a nuclear safety concern.

4.7.B/4.7.C Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip logic demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary

3.7/4.7 BASES (Cont'd)

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.



3.7/4.7 BASES (Cont'd)

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the operable systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contain the list of all the Primary Containment Isolation Valves for which the TS requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. Section 1.0, Definitions, paragraph 0.1 allows normally locked and sealed-closed Primary Containment Isolation Valves to be opened to perform necessary operational activities under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated when a valid primary containment isolation signal is indicated.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level ($\geq 398"$) in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at $\geq 398"$ or main steam line high radiation.

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

3.7/4.7 BASES (Cont'd)

in the system, isolation is provided by high temperature in the cleanup system area or high drain temperature. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - (Deleted)

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

3.7/4.7 BASES (Cont'd)

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip logic are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability in accordance with Specification 1.0.MM results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested per Specification 1.0.MM to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25-inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage. During cycle 6, CREVS has been declared inoperable only because it does not meet its design basis for essentially zero unfiltered inleakage. Reactor power operations and fuel movement are acceptable until just prior to startup for unit 2 cycle 7. During cycle 6, CREVS must be demonstrated to be functional by performing all applicable surveillances. In the event that the applicable surveillances are not successfully performed, the actions required by the LCOs must be complied with.

High efficiency particulate absolute (HEPA) filters are installed prior to the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a

3.7/4.7 BASES (Cont'd)

limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to Cold Shutdown within 24 hours or refueling operations are terminated.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with USAEC Report-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

3.7.F/4.7.F Primary Containment Purge System

The Primary Containment Purge System is a non-safety related system that is normally isolated and normally not required to be functional during power operation. The system is designed to provide the preferred exhaust path for purging the primary containment system; however, the Standby Gas Treatment System can be used to perform the equivalent function.

3.7/4.7 BASES (Cont'd)

When the Primary Containment Purge System is in operation, the exhaust from the primary containment is first processed by a filter train assembly and then channeled through the reactor building roof exhaust system.

The filter train assembly contains a HEPA (high efficiency particulate air) filter, charcoal adsorber, and centrifugal fan. In-place tests are performed to ensure leak tightness of the filter train assembly of at least 99% and a HEPA efficiency of at least 99% removal of DOP particulates. Laboratory tests are performed on adsorber carbon samples to ensure an 85% removal efficiency for radioactive methyl iodide. Tests are performed to ensure that the system is not operating at a flow significantly different from the design flow, which may affect the removal efficiency of the HEPA filters and charcoal adsorbers. The pressure drop across the combined HEPA filters and charcoal adsorbers is checked once per operating cycle to be less than 8.5 inches of water at the system design flow rate to ensure that the filters and adsorbers are not clogged with excessive amounts of foreign matter.

The above tests are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with USAEC Report-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

If significant painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

The primary containment isolation valves associated with the purging of the primary containment are FCV 64-17, 64-18, 64-19, 64-29, 64-30, 64-32, 64-33, and 76-24. These valves may be open for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. Closure of these large diameter valves within the 24-hour period is needed to

3.7/4.7 BASES (Cont'd)

retain the reduced oxygen concentration specified in Technical Specification 3.7.A.5.b, and to minimize the time period which the primary containment is not isolated per the guidelines of Branch Technical Position CSB 6-4.

When the large diameter valves noted above are closed, primary containment venting is performed using valves FCV 64-31, 64-34, and 84-20 and the Standby Gas Treatment System. The operability of these primary containment isolation valves is governed by Technical Specification 3.7.D. The operability of the Standby Gas Treatment System is governed by Technical Specification 3.7.B.

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BFN
Unit 3

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
1(9)	Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	≤ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate refueling floor c. Close atmosphere control system.
2(7) (8)	Instrument Channel SGTS Flow - Train A R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train B R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train C R. H. Heaters	≥ 2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
1	Reactor Building Isolation Timer (refueling floor)	$0 \leq t \leq 2$ secs.	H or F	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
1	Reactor Building Isolation Timer (reactor zone)	$0 \leq t \leq 2$ secs.	G or A or H	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
2(10)	Group 1 (Initiating) Logic	N/A	A	1. Group 1: The valves in Group 1 are actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure

3.2/4.2-9

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

BPN Unit 3	Minimum No. Instrument Channels Operable per Trip Sys(1)(1)	Function	Trip Level Setting	Action (1)	Remarks
3.2/4.2-10	1	Group 1 (Actuation) Logic	N/A	B	1. Group 1: The valves in Group 1 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure
	2	Group 2 (Initiating) Logic	N/A	A or (B and E)	1. Group 2: The valves in Group 2 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure
	1	Group 2 (RHR Isolation-Actuation) Logic	N/A	D	
	1	Group 8 (TIP-Actuation) Logic	N/A	J	
	1	Group 2 (Drywell Sump Drains-Actuation) Logic	N/A	K	
	1	Group 2 (Reactor Building & Refueling Floor, and Drywell Vent and Purge-Actuation) Logic	N/A	F and G	1. Part of Group 6 Logic
	2	Group 3 (Initiating) Logic	N/A	C	1. Group 3: The valves in Group 3 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Reactor Water Cleanup System High Temperature c. Reactor Water Cleanup System High Drain Temperature



TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
1	Group 3 (Actuation) Logic	N/A	C	
1	Group 6 Logic	N/A	F and G	1. Group 6: The valves in Group 6 are actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure c. Reactor Building Ventilation High Radiation
1	Group 8 (Initiating) Logic	N/A	J	1. Group 8: The valves in Group 8 are automatically actuated by only the following conditions: a. High Drywell Pressure b. Reactor Vessel Low Water Level 2. Same as Group 2 initiating logic.
1	Reactor Building Isolation (refueling floor) Logic	N/A	H or F	
1	Reactor Building Isolation (reactor zone) Logic	N/A	H or G or A	
1(7) (8)	SGTS Train A Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train B Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train C Logic	N/A	L or (A and F)	

Refer to Table 3.2.B for RCIC and HPCI functions including Groups 4, 5, and 7 valves.

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Unit 3

3.2/4.2-11a |

NOTES FOR TABLE 3.2.A (Cont'd)

4. Only required in RUN MODE (interlocked with Mode Switch).
5. Deleted
6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.
7. A train is considered a trip system.
8. Two out of three SGTS trains required. A failure of more than one will require actions A and F.
9. DELETED
10. DELETED
11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
12. A channel contains four sensors, all of which must be OPERABLE for the channel to be OPERABLE.

Power operations permitted for up to 30 days with 15 of the 16 temperature switches OPERABLE.

In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves.

13. The nominal setpoints for alarm and reactor trip (1.5 and 3.0 times background, respectively) are established based on the normal background at full power. The allowable setpoints for alarm and reactor trip are 1.2-1.8 and 2.4-3.6 times background, respectively.
14. Requires two independent channels from each physical location; there are two locations.

TABLE 3.2.B (Continued)

Unit	BFN	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
3		2(2)	Instrument Channel - Reactor High Water Level	≤583" above vessel zero.	A	1. Above trip setting trips HPCI turbine.
		1	Instrument Channel - HPCI Turbine Steam Line High Flow	≤90 psi (7)	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
3.2/4.2-19		4(4)	Instrument Channel - HPCI Steam Line Space High Temperature	≤200°F.	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
		3(2)	Instrument Channel - HPCI Steam Supply Pressure - Low (PS 73-1A-D)	≥100 psig	A	1. Below trip setting isolates HPCI system and trips HPCI turbine.
		3(2)	Instrument Channel - HPCI Turbine Exhaust Diaphragm (PS 73-20A-D)	≤20 psig	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
		1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
		1	RCIC System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. 2. Group 7: The valves in Group 7 are automatically actuated by only the following condition: 1. The respective turbine steam supply valve not fully closed.
	1	RCIC System (Isolation) Logic	N/A	B	1. Includes Group 5 valves. 2. Group 5: The valves in Group 5 are actuated by any of the following conditions: a. RCIC Steamline Space High Temperature b. RCIC Steamline High Flow c. RCIC Steamline Low Pressure d. RCIC Turbine Exhaust Diaphragm High Pressure	
		1 (16)	ADS Logic	N/A	A	

TABLE 3.2.B (Continued)

Unit	BPN	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
3.2/4.2-20		1	RHR (LPCI) System (Initiation)	N/A	B	
		1	RHR (LPCI) System (Containment Cooling Spray) Logic	N/A	A	
		1	HPCI System (Initiating) Logic	N/A	B	<ol style="list-style-type: none"> Includes Group 7 valves. Group 7: The valves in Group 7 are automatically actuated by only the following condition: <ol style="list-style-type: none"> The respective turbine steam supply valve not fully closed.
		1	HPCI System (Isolation) Logic	N/A	B	<ol style="list-style-type: none"> Includes Group 4 valves. Group 4: The valves in Group 4 are actuated by any of the following conditions: <ol style="list-style-type: none"> HPCI Steamline Space High Temperature HPCI Steamline High Flow HPCI Steamline Low Pressure HPCI Turbine Exhaust Diaphragm High Pressure
		1(3)	Core Spray Loop A Discharge Pressure (PI-75-20)	0 - 500 psig Indicator (9)	D	<ol style="list-style-type: none"> Part of filled discharge pipe requirements. Refer to Section 4.5.
		1(3)	Core Spray Loop B Discharge Pressure (PI-75-48)	0 - 500 psig Indicator (9)	D	<ol style="list-style-type: none"> Part of filled discharge pipe requirements. Refer to Section 4.5.
		1(3)	RHR Loop A Discharge Pressure (PI-74-51)	0 - 450 psig Indicator (9)	D	<ol style="list-style-type: none"> Part of filled discharge pipe requirements. Refer to Section 4.5.
		1(3)	RHR Loop B Discharge Pressure (PI-74-65)	0 - 450 psig Indicator (9)	D	<ol style="list-style-type: none"> Part of filled discharge pipe requirements. Refer to Section 4.5.
		1(10)	Instrument Channel - RHR Start	N/A	A	<ol style="list-style-type: none"> Starts RHR area cooler fan when respective RHR motor starts.

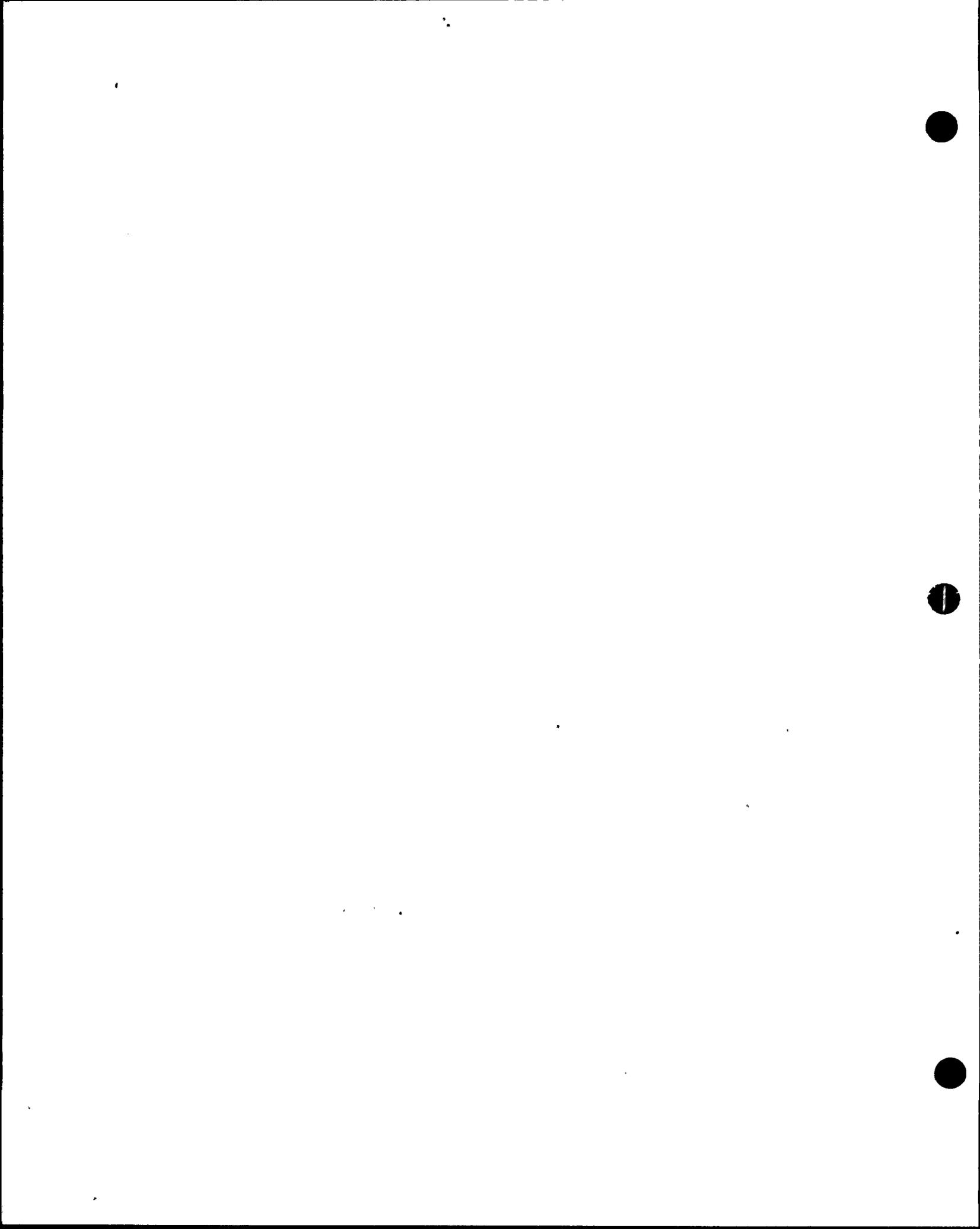


TABLE 3.2.B (Continued)

BFN Unit 3	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
	1(10)	Instrument Channel - Thermostat (RHR Area Cooler Fan)	≤100°F	A	1. Above trip setting starts RHR area cooler fans.
	2(10)	Instrument Channel - Core Spray A or C Start	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts.
	2(10)	Instrument Channel - Core Spray B or D	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts.
	1(10)	Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	≤ 100°F	A	1. Above trip setting starts Core Spray area cooler fans.
	1(10)	RHR Area Cooler Fan Logic	N/A	A	
	1(10)	Core Spray Area Cooler Fan Logic	N/A	A	
	1(11)	Instrument Channel - Core Spray Motors A or C Start	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
	1(11)	Instrument Channel - Core Spray Motors B or D Start	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
	1(12)	Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
	1(12)	Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
	1	RPT Logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.
	1(13)	RHRSW Initiate Logic	N/A	(14)	

3.2/4.2-21

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at 378 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). These trip settings are adequate to prevent core uncovering in the case of a break in the largest line assuming the maximum closing time. †

4.7.A. Primary Containment

4.7.A.2.g (Cont'd)

The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

+

3.7.C. Secondary Containment

3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

D. Primary Containment Isolation Valves

1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation



3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.7.D. Primary Containment Isolation Valves

2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:
 - a. The inoperable valve is restored to OPERABLE status, or
 - b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.
3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.D. Primary Containment Isolation Valves

4.7.D.1.a (Cont'd)

- and in accordance with Specification 1.0.MM, tested for closure times.
- b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested
 - c. (Deleted)
 - d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.
2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occurring.

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49.6 psig, P_a . As an added conservatism, the measured overall integrated leakage rate is further limited to $0.75 L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of three feet seven inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately three feet and water volume of approximately 123,000 cubic feet.

3.7/4.7 BASES (Cont'd)

Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. The maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and loss-of-coolant accident (LOCA) cases. The actions required by Specifications 3.7.C.-3.7.F. assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for Core Standby Cooling Systems operability. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for controlled blowdown anytime during RCIC operation and assures margin for complete condensation of steam from the design basis loss-of-coolant accident.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve; (2) initiate suppression pool water cooling heat exchangers; (3) initiate reactor shutdown; and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.



3.7/4.7 BASES (Cont'd)

In conjunction with the Mark I Containment Short Term Program, a plant-unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.1 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.06 feet to 3.58 feet will assure the integrity of the suppression chamber when subjected to post-loss-of-coolant suppression pool hydrodynamic forces.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain low leakage integrity. The <4 percent hydrogen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged sufficient to perform the leak inspection and establish the required oxygen concentration.

To ensure that the hydrogen concentration is maintained less than 4 percent following an accident, liquid nitrogen is maintained onsite for containment atmosphere dilution. About 2,260 gallons would be sufficient as a seven-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2,500 gallons is conservative. Following a loss-of-coolant accident the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation. Failure of one system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

3.7/4.7 BASES (Cont'd)

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post-loss-of-coolant accident conditions. Post-loss-of-coolant accident calculations show that the CAD system within two hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.9-percent (at 3 hours) and 3.9-percent (at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism (\leq 3.9-percent), as a guide for CAD/Purge operations.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100-percent vacuum relief breakers (two parallel sets of two valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One reactor building vacuum breaker may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle the position indicating lights in the control room are designed to function as specified below:

Initial and Final Condition	Check - On	(Fully Closed)
	Green - On	
	Red - Off	
Opening Cycle	Check - Off	(Cracked Open)
	Green - Off	(> 80° Open)
	Red - On	(> 3° Open)
Closing Cycle	Check - On	(Fully Closed)
	Green - On	(< 80° Open)
	Red - Off	(< 3° Open)

The valve position indicating lights consist of one check light on the check light panel which confirms full closure, one green light next to the hand switch which confirms 80° of full opening and one red light next to the hand switch which confirms "near closure" (within 3° of full closure). Each light is on a separate switch. If the check light circuit is operable when the valve is exercised by its air operator there exists a confirmation that the valve will fully close. If the red light circuit is operable, there exists a

3.7/4.7 BASES (Cont'd)

confirmation that the valve will at least "nearly close" (within 3° of full closure). The green light circuit confirms the valve will fully open. If none of the lights change indication during the cycle, the air operator must be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully operable. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered operable the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten operable to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure and held constant. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

3.7/4.7 BASES (Cont'd)

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a LOCA. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this Bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

3.7/4.7 BASES (Cont'd)

Establishing the test limit of 2.0-percent/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate by 0.75 thereby providing a 25-percent margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 10 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, determining the oxygen concentration twice a week serves as an added assurance that the oxygen concentration will not exceed 4 percent.

3.7.B/3.7.C Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment, if required, when the reactor is shutdown and the drywell is open. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. All three standby gas treatment system fans are designed to

3.7/4.7 BASES (Cont'd)

automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage.

High efficiency particulate air (HEPA) filters are installed before and after the charcoal absorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine absorbers. The charcoal absorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal absorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal absorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal absorbers.

Only two of the three standby gas treatment systems are needed to clean up the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If more than one train is inoperable, all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel must be suspended and all reactors placed in a cold shutdown condition, because the remaining train would provide only 50 percent of the capacity required to filter and exhaust the reactor building atmosphere to the stack. Suspension of these activities shall not preclude movement of a component to a safe, conservative position. Operations that have the potential for draining the reactor vessel must be suspended as soon as practical to minimize the probability of a vessel draindown and subsequent potential for fission product release. Draindown of a reactor vessel containing no fuel does not present the possibility for fuel damage or significant fission product release and therefore is not a nuclear safety concern.

4.7.B/4.7.C Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip logic demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary

3.7/4.7 BASES (Cont'd)

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

3.7/4.7 BASES (Cont'd)

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the operable systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contain the list of all the Primary Containment Isolation Valves for which the TS requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. Section 1.0, Definitions, paragraph 0.1 allows normally locked and sealed-closed Primary Containment Isolation Valves to be opened to perform necessary operational activities under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated when a valid primary containment isolation signal is indicated.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level (378") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at 378" or main steam line high radiation.

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - Process lines are closed only on the respective turbine steam supply valve not fully closed. This assures that the valves are not open when HPCI or RCIC action is required.

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

3.7/4.7 BASES (Cont'd)

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip logic are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability in accordance with Specification 1.0.MM results in a greater assurance that the valve will be operable when needed.

The main steamline isolation valves are functionally tested per Specification 1.0.MM to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25-inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage. During cycle 6, CREVS has been declared inoperable only because it does not meet its design basis for essentially zero unfiltered inleakage. Reactor power operations and fuel movement are acceptable until just prior to startup for unit 2 cycle 7. During cycle 6, CREVS must be demonstrated to be functional by performing all applicable surveillances. In the event that the applicable surveillances are not successfully performed, the actions required by the LCOs must be complied with.

High efficiency particulate absolute (HEPA) filters are installed prior to the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a

3.7/4.7 BASES (Cont'd)

limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to Cold Shutdown within 24 hours or refueling operations are terminated.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with USAEC Report-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

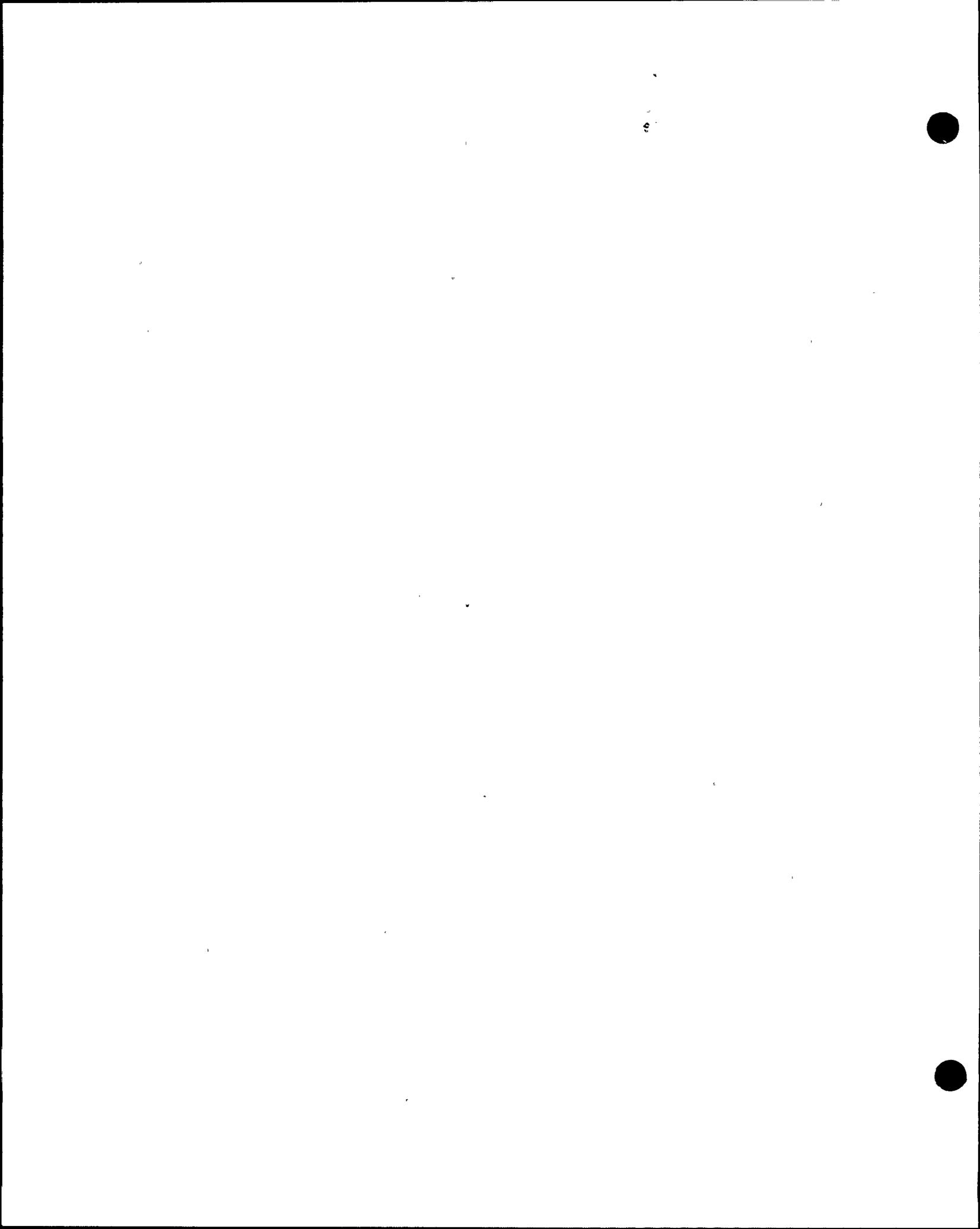
Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

3.7.F/4.7.F Primary Containment Purge System

The Primary Containment Purge System is a non-safety related system that is normally isolated and normally not required to be functional during power operation. The system is designed to provide the preferred exhaust path for purging the primary containment system; however, the Standby Gas Treatment System can be used to perform the equivalent function.



3.7/4.7 BASES (Cont'd)

When the Primary Containment Purge System is in operation, the exhaust from the primary containment is first processed by a filter train assembly and then channeled through the reactor building roof exhaust system.

The filter train assembly contains a HEPA (high efficiency particulate air) filter, charcoal adsorber, and centrifugal fan. In-place tests are performed to ensure leak tightness of the filter train assembly of at least 99% and a HEPA efficiency of at least 99% removal of DOP particulates. Laboratory tests are performed on adsorber carbon samples to ensure an 85% removal efficiency for radioactive methyl iodide. Tests are performed to ensure that the system is not operating at a flow significantly different from the design flow, which may affect the removal efficiency of the HEPA filters and charcoal adsorbers. The pressure drop across the combined HEPA filters and charcoal adsorbers is checked once per operating cycle to be less than 8.5 inches of water at the system design flow rate to ensure that the filters and adsorbers are not clogged with excessive amounts of foreign matter.

The above tests are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with USAEC Report-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

If significant painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

The primary containment isolation valves associated with the purging of the primary containment are FCV 64-17, 64-18, 64-19, 64-29, 64-30, 64-32, 64-33, and 76-24. These valves may be open for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. Closure of these large diameter valves within the 24-hour period is needed to

3.7/4.7 BASES (Cont'd)

retain the reduced oxygen concentration specified in Technical Specification 3.7.A.5.b, and to minimize the time period which the primary containment is not isolated per the guidelines of Branch Technical Position CSB 6-4.

When the large diameter valves noted above are closed, primary containment venting is performed using valves FCV 64-31, 64-34, and 84-20 and the Standby Gas Treatment System. The operability of these primary containment isolation valves is governed by Technical Specification 3.7.D. The operability of the Standby Gas Treatment System is governed by Technical Specification 3.7.B.

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BFN
Unit 3

3.7/4.7-39 |

ENCLOSURE 2

DESCRIPTION AND JUSTIFICATION BROWNS FERRY NUCLEAR PLANT (BFN)

REASON FOR CHANGE

BFN Units 1, 2 and 3 Technical Specifications (TS) are being revised to remove component lists and references to the component lists from the TS in accordance with Generic Letter 91-08, "Removal of Component Lists from Technical Specifications (Generic Letter 91-08)". This generic letter provides an acceptable alternative to identifying every component by its plant identification number as it is currently listed in the tables of TS components. The removal of component lists is acceptable because it does not alter existing TS requirements or those components to which they apply.

DESCRIPTION OF THE PROPOSED CHANGE

Note: Changes apply to all units unless specified otherwise.

1. Existing TS

The existing TS Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, Function Column, "Group 1 (Initiating) Logic - Remarks 1, " page 3.2/4.2-9 states: "Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1 and add the following Group 1 valve actuation remarks:

- "1. Group 1: The valves in Group 1 are actuated by any of the following conditions:
1. Reactor Vessel Low Low Water Level
 2. Main Steamline High Radiation
 3. Main Steamline High Flow
 4. Main Steamline Space-High Temperature
 5. Main Steamline Low Pressure"



2. Existing TS

The existing TS Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, Function Column, "Group 1 (Actuation) Logic - Remarks 1, " page 3.2/4.2-10 states: "Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1 and add the following Group 1 valve actuation remarks:

"1. Group 1: The valves in Group 1 are actuated by any of the following conditions:

1. Reactor Vessel Low Low Water Level
2. Main Steamline High Radiation
3. Main Steamline High Flow
4. Main Steamline Space High Temperature
5. Main Steamline Low Pressure"

3. Existing TS

The existing TS Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, Function Column, "Group 2 (Initiating) Logic - Remarks 1, " page 3.2/4.2-10 states: "Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1 and add the following Group 2 valve actuation remarks:

"1. Group 2: The valves in Group 2 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level
2. High Drywell Pressure"

4. Existing TS

The existing Units 1 and 3 TS Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, Function Column, "Group 3 (Initiating) Logic - Remarks 1, " page 3.2/4.2-10 states: "Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1 and add the following Group 3 valve actuation remarks:

"1. Group 3: The valves in Group 3 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level
2. Reactor Water Cleanup System High Temperature
3. Reactor Water Cleanup System High Drain Temperature"

5. Existing TS

The existing Unit 2 TS Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, Function Column, "Group 3 (Initiating) Logic - Remarks 1, " page 3.2/4.2-10 states: "Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1 and add the following Group 3 valve actuation remarks:

"1. Group 3: The valves in Group 3 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level
2. Reactor Water Cleanup (RWCU) System High Temperature in the main steam valve vault,
3. RWCU System High Temperature in RWCU pump room 2A,
4. RWCU System High Temperature in the RWCU pump room 2B,
5. RWCU System High Temperature in RWCU heat exchanger room,
6. RWCU System High Temperature in the space near the pipe trench containing RWCU piping."



6. Existing TS

The existing TS Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, Function Column, "Group 6 Logic - Remarks 1, " page 3.2/4.2-10 states: "Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1 and add the following Group 6 valve actuation remarks:

"1. Group 6: The valves in Group 6 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level
2. High Drywell Pressure
3. Reactor Building Ventilation High Radiation

7. Existing TS

The existing TS Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation, Function Column, "Group 8 (Initiating) Logic - Remarks 1, " page 3.2/4.2-10 states: "Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1 and add the following Group 8 valve actuation remarks:

"1. Group 8: The valves in Group 8 are automatically actuated by only the following conditions:

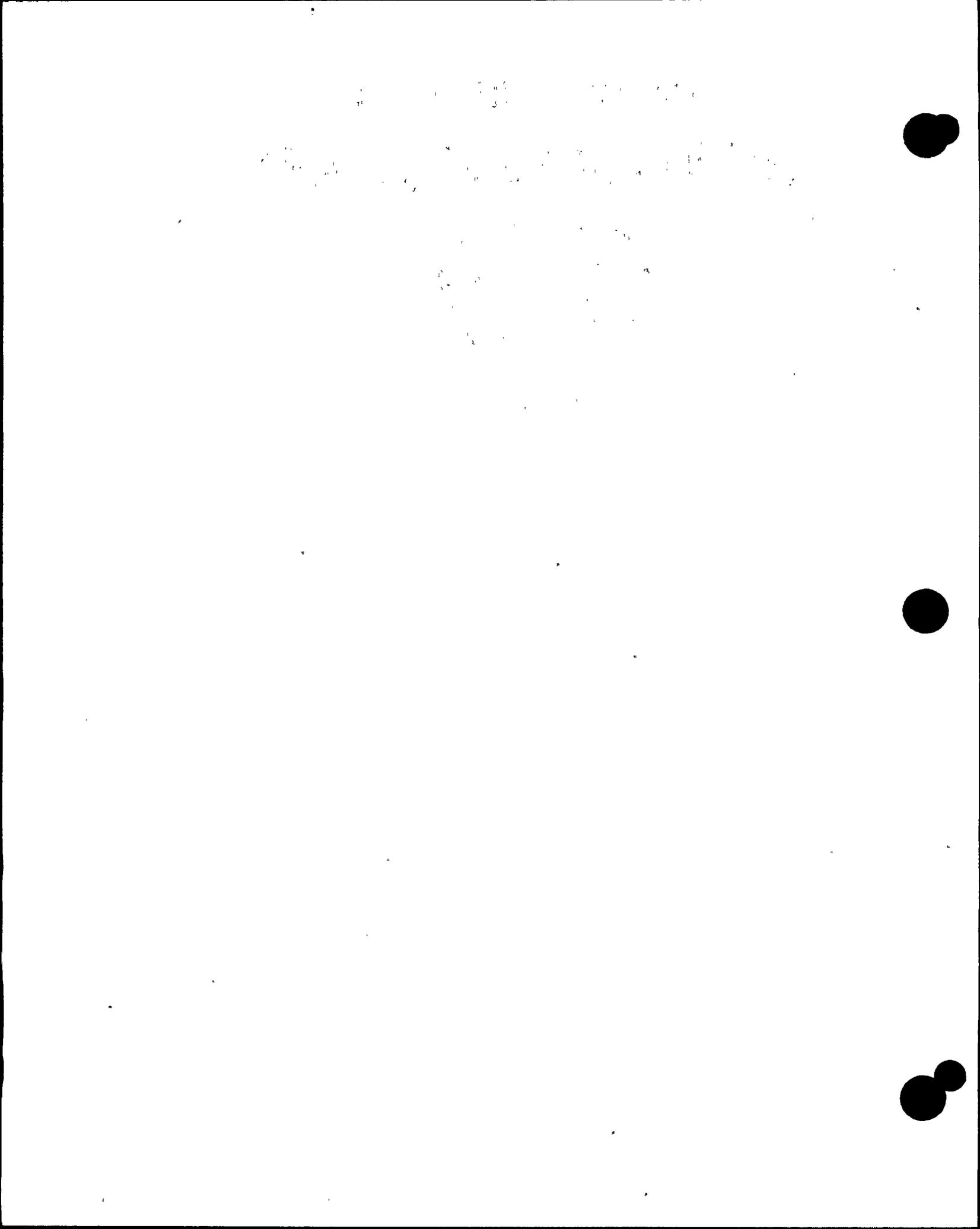
1. High Drywell Pressure
2. Reactor Vessel Low Water Level"

8. Existing TS

The existing TS Table 3.2.A, Notes for Table 3.2.A, item 10, page 3.2/4.2-13 states: "Refer to Table 3.7.A and its notes for a listing of Isolation Valve Groups and their initiating signals."

Proposed Change

The proposed change would delete item 10 from the Notes for Table 3.2.A.



9. Existing TS

The existing Units 1 and 3 TS Table 3.2.B, Instrumentation That Initiates or Controls the Core and Containment Cooling Systems, Function column, "RCIC System (Initiating) Logic - Remarks 1, page 3.2/4.2-19 states: "Includes Group 7 valves. Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the sentence, "Refer to Table 3.7.A for list of valves." and add the following Group 7 valve actuation remarks:

- "2. Group 7: The valves in Group 7 are automatically actuated by only the following condition:
 1. The respective turbine steam supply valve not fully closed."

10. Existing TS

The existing Unit 2 TS Table 3.2.B, Instrumentation That Initiates or Controls the Core and Containment Cooling Systems, Function column, "RCIC System (Initiating) Logic - Remarks 1, page 3.2/4.2-19 states: "Includes Group 7 valves. Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1.

11. Existing TS

The existing TS Table 3.2.B, Instrumentation That Initiates or Controls the Core and Containment Cooling Systems, Function column, "RCIC System (Isolation) Logic - Remarks 1," page 3.2/4.2-19 states: "Includes Group 5 valves. - Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the sentence, "Refer to Table 3.7.A for list of valves" and add the following Group 5 valve actuation remarks:

- "2. Group 5: The valves in Group 5 are actuated by any of the following conditions:
 1. RCIC Steamline Space High Temperature
 2. RCIC Steamline High Flow
 3. RCIC Steamline Low Pressure
 4. RCIC Turbine Exhaust Diaphragm High Pressure"



12. Existing TS

The existing Units 1 and 3 TS Table 3.2.B, Instrumentation That Initiates or Controls the Core and Containment Cooling Systems, Function column, "HPCI System (Initiating) Logic - Remarks 1," page 3.2/4.2-20 states: "Includes Group 7 valves. - Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the sentence "Refer to Table 3.7.A for list of valves." and add the following Group 7 valve actuation remarks:

- "2. Group 7: The valves in Group 7 are automatically actuated by only the following condition:
 1. The respective turbine steam supply valve not fully closed."

13. Existing TS

The existing Unit 2 TS Table 3.2.B, Instrumentation That Initiates or Controls the Core and Containment Cooling Systems, Function column, "HPCI System (Initiating) Logic - Remarks 1," page 3.2/4.2-20 states: "Includes Group 7 valves. - Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the remarks column, item 1.

14. Existing TS

The existing TS Table 3.2.B, Instrumentation That Initiates or Controls the Core and Containment Cooling Systems, Function column, "HPCI System (Isolation) Logic - Remarks 1, page 3.2/4.2-20 states: "Includes Group 4 valves. - Refer to Table 3.7.A for list of valves."

Proposed Change

The proposed change would delete the sentence, "Refer to Table 3.7.A for list of valves" and add the following Group 4 valve actuation remarks:

- "2. Group 4: The valves in Group 4 are actuated by any of the following conditions:
 1. HPCI Steamline Space High Temperature
 2. HPCI Steamline High Flow
 3. HPCI Steamline Low Pressure
 4. HPCI Turbine Exhaust Diaphragm High Pressure"



15. Existing TS

The existing TS Section 3.2/4.2 Bases, paragraph 6, pages 3.2/4.2-65 for Units 1 and 2, and paragraph 6, page 3.2/4.2-64 for Unit 3 contain the following sentence: "Details of valve grouping and required closing times are given in Specification 3.7." This sentence is a reference to the components lists.

Proposed Change

The proposed change would delete the sentence from TS Section 3.2/4.2, Bases, paragraph 6 for all three units.

16. Existing TS

The existing Units 1 and 3 TS Surveillance Requirements for Primary Containment Isolation Valves, Section 4.7.A.2.g, page 3.7/4.7-7 states: "Penetrations and isolation valves are identified as follows:

- (1) Testable penetrations with double O-ring seals - Table 3.7.B,
- (2) Testable penetrations with testable bellows - Table 3.7.C,
- (3) Isolation valves without fluid seal - Table 3.7.D,
- (4) Testable electrical penetrations - Table 3.7.H, and
- (5) Isolation valves sealed with fluid - Tables 3.7.E, and 3.7.F".

Proposed Change

The proposed change would delete that portion of Section 4.7.A.2g that begins with "Penetrations and isolation valves are identified as follows: . . ." and includes items (1) through (5).

17. Existing TS

The existing TS Limiting Conditions for Operation for Primary Containment Isolation Valves, Section 3.7.D.1, page 3.7/4.7-17 states: "When Primary Containment Integrity is required, all isolation valves listed in Table 3.7.A and all reactor coolant system instrument lines flow check valves shall be OPERABLE except as specified in 3.7.D.2."

Proposed Change

The proposed change would revise Section 3.7.D.1 to read: "When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2."

18. Existing TS

The existing TS Surveillance Requirements for Primary Containment Isolation Valves, Section 4.7.D.1.a, page 3.7/4.7-17 states: "At least once per operating cycle, the OPERABLE isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and in accordance with Specification 1.0.MM, tested for closure times."

Proposed Change

The proposed change would revise Section 4.7.D.1.a to read: "At least once per operating cycle the OPERABLE primary containment isolation valves . . ."

19. Existing TS

The existing TS Limiting Conditions for Operations for Primary Containment Isolation Valve, Section 3.7.D.2, page 3.7/4.7-18 states: "In the event any isolation valve specified in Table 3.7.A becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either: . . ."

Proposed Change

The proposed change would revise Section 3.7.D.2 to read: "In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either: . . ."

20. Existing TS

The existing TS Surveillance Requirements for Primary Containment Isolation Valves, Section 4.7.D.1.b, page 3.7/4.7-18 states: "In accordance with Specification 1.0.MM, all normally open power operated isolation valves shall be functionally tested."

Proposed Change

The proposed change would revise Section 4.7.D.1.b, to read: "In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested."

21. Existing TS

The existing TS Surveillance Requirements for Primary Containment Isolation Valves, Section 4.7.D.2, page 3.7/4.7-18 states:

"Whenever an isolation valve listed in Table 3.7.A is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily."

Proposed Change

The proposed change would revise Section 4.7.D.2 to read:

"Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily."

22. Existing TS

The existing TS Table 3.7.A, Primary Isolation Valves is a listing of components by its plant identification number. Pages 3.7/4.7-25 through 3.7/4.7-29 for Unit 1, pages 3.7/4.7-25 through 3.7/4.7-33 for Unit 2, and pages 3.7/4.7-24 through 3.7/4.7-28 for Unit 3 contains the Primary Containment Isolation Valves listing. Notes for Table 3.7.A are contained on pages 3.7/4.7-30 through 3.7/4.7-31 for Unit 1, pages 3.7/4.7-34 through 3.7/4.7-35 for Unit 2, and pages 3.7/4.7-29 through 3.7/4.7-30 for Unit 3.

Proposed Change

The proposed change would delete Table 3.7.A and the notes for Table 3.7.A for all three units.

23. Existing TS

The existing Units 1 and 3 TS Table 3.7.B, Testable Penetrations with Double O-Ring seals is a listing of TS components by its plant identification number. Page 3.7/4.7-32 for Unit 1 and page 3.7/4.7-31 for Unit 3 contains the list of Testable Penetrations with double O-Ring Seals.

Proposed Change

The proposed change would delete Table 3.7.B from Units 1 and 3 TS.

24. Existing TS

The existing Units 1 and 3 TS Table 3.7.C, Testable Penetrations with Testable Bellows is a listing of components by its plant identification number. Page 3.7/4.7-33 for Unit 1 and page 3.7/4.7-32 for Unit 3 contains the list of Testable Penetrations with Testable Bellows.

Proposed Change

The proposed change would delete Table 3.7.C from Units 1 and 3 TS.



25. Existing TS

The existing Units 1 and 3 TS Table 3.7.D, Air Tested Isolation Valves is a listing of components by its plant identification number. Pages 3.7/4.7-34 through 3.7/4.7-36 for Unit 1 and pages 3.7/4.7-33 through 3.7/4.7-35 for Unit 3 contains the list of Air Tested Isolation Valves.

Proposed Change

The proposed change would delete Table 3.7.D from Units 1 and 3 TS.

26. Existing TS

The existing Units 1 and 3 TS Table 3.7.E, Primary Containment Isolation Valves Which Terminate Below The Suppression Pool Water Level is a listing of TS components by its plant identification number. Page 3.7/4.7-37 for Unit 1 and page 3.7/4.7-36 for Unit 3 contains the list of primary containment isolation valves which terminate below the suppression pool water level.

Proposed Change

The proposed change would delete Table 3.7.E from the Units 1 and 3 TS.

27. Existing TS

The existing Units 1 and 3 TS Table 3.7.F, Primary Containment Isolation Valves Located In Water Sealed Seismic Class 1 Lines is a listing of TS components by its plant identification number. Page 3.7/4.7-38 for Unit 1 and page 3.7/4.7-37 for Unit 3 contains the list of primary containment isolation valves located in water sealed seismic Class 1 lines.

Proposed Change

The proposed change would delete Table 3.7.F from the Units 1 and 3 TS.

28. Existing TS

The existing Units 1 and 3 TS Table 3.7.H, Testable Electrical Penetrations is a listing of TS components by its plant identification number. Pages 3.7/4.7-39 through 3.7/4.7-40 for Unit 1 and page 3.7/4.7-38 for Unit 3 contains the list of testable electrical penetrations.

Proposed Change

The proposed change would delete Table 3.7.H from the Units 1 and 3 TS.

29. Proposed Change

The proposed change would revise TS Section 3.7/4.7 Bases, paragraph 3.7.D/4.7.D, Primary Containment Isolation Valves, page 3.7/4.7-49 for Unit 1; page 3.7/4.7-44 for Unit 2; and page 3.7/4.7-47 for

Unit 3 by adding the following paragraph under the heading:

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contain the list of all the Primary Containment Isolation Valves for which the TS requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. Section 1.0, Definitions, paragraph 0.1 allows normally locked and sealed-closed Primary Containment Isolation Valves to be opened to perform necessary operational activities under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated when a valid primary containment isolation signal is indicated.

JUSTIFICATION FOR CHANGE

1. NRC Generic Letter 91-08, Removal of Components Lists from Technical Specifications (TS) dated May 6, 1991 allows the licensee to remove component lists from TS. GL 91-08 is an acceptable alternative to identifying every component by its plant identification number as it is currently listed in Tables 3.7.A through 3.7.F and 3.7.H of the BFN TS. The removal of these component lists does not alter existing TS requirements or those components to which they apply.
2. Tables 3.7.A through 3.7.F and 3.7.H will be incorporated into BFN plant procedures that are subject to the change control provisions for plant procedures in the administrative controls section of the TS. The removal of component lists from the TS permits administrative control of changes to these lists without processing a license amendment. Any change to component lists contained in plant procedures is subject to the requirements specified in Chapter 6 of the BFN TS. Therefore, the change control provisions of the TS provide an adequate means to control changes to these component lists, when they have been incorporated into plant procedures.
3. The definition of Unit 2, Group 3 valves in the notes for Table 3.7.A, differs from Units 1 and 3. The reason for the difference was contained in Amendment 189 wherein modifications were made to delete the non-environmentally qualified (EQ) temperature switches used to detect RWCU line breaks and replace them with EQ resistance temperature detectors. Consequently, changes were made to the TS to reflect the modifications. The changes were subsequently included in the Table 3.7.A actuation remarks for Group 3 valves.

There were other differences in Unit 2 TS for the 3.7 tables such as the removal of Tables 3.7.B through 3.7.F and 3.7.H from the Unit 2 TS. Both Units 1 and 3 retained the tables, however, the Unit 2 tables were deleted by TS amendment 193. Tables 3.7.B, C, and H were removed because the information was contained in the BFN Appendix J program. Tables 3.7.D, E, and F information were combined with Table 3.7.A.

Group 7 valves actuation remarks were removed from the Unit 2 TS Notes for Table 3.7.A and the 3.7/4.7 Bases by Amendment 193, however the Group 7 valves references were left in Table 3.2.B and the 3.2 Bases on page 3.2/4.2-65 by Amendment 193. This was an administrative error that the TS 297 submittal will correct. The Group 7 actuation remarks remain for Units 1 and 3 in Table 3.2.B and the 3.7/4.7 Bases.

4. The portions of the Limiting Conditions for Operation (LCO), Surveillance Requirements, and other tables that reference the components lists will be removed to support the justification for change discussed in 1 through 3 above.
5. The TS bases are updated to describe the methodology used to remove the components lists from the TS and incorporate the lists in plant procedures under the guidance provided by Generic Letter 91-08.

11/11/11

SUMMARY OF GENERIC LETTER 91-08 REQUIREMENTS TO REMOVE COMPONENT LISTS FROM TS

Generic Letter 91-08 allows TVA/BFN to submit a request to remove component lists from TS and by addressing the following issues:

1. Each TS should include an appropriate description of the scope of the components to which the TS requirements apply. Components that are defined by regulatory requirements or guidance need not be clarified further. However, the Bases Section of the TS should reference the applicable requirements or guidance.

TVA/BFN Response: The current TS Limiting Condition for Operation (LCO) requires all isolation valves to be operable and this meets the intent of the description of the components to which the TS requirements apply. There were no components identified that are defined by regulatory requirements or guidance.

2. If the removal of a component list results in the loss of notes that modify or provide an exception to the TS requirements, the specification should be revised to incorporate that modification or exception. The modification or exception should be stated in terms that identify any group of components by function rather than by plant identification number, if practical.

TVA/BFN Response: The loss of notes is not applicable since all notes are being incorporated in the BFN procedures and they do not contain exceptions to the TS requirements.

3. Licensees should confirm that the lists of components removed from the TS are located in appropriately controlled plant procedures. The list of components may be included in the next update of the FSAR. The Bases Section of individual specifications also may reference the plant procedures or other documents that identify each component list. The Bases Section for the containment isolation valve TS should be updated to describe the intent of opening valves under administrative control.

TVA/BFN Response: The present TS Section 6.8.1.1.a, which covers procedures required by Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, includes the plant procedure that controls the primary containment isolation valve listing. The list of components will not be included in the FSAR or the Bases Section of individual specifications. However, the Bases Section has been updated to describe the intent of opening valves under administrative control.

