

UNIT 1
EFFECTIVE PAGE LIST

REMOVE

3.2/4.2-14
3.2/4.2-15
3.2/4.2-23
3.2/4.2-24
3.5/4.5-7
3.5/4.5-8
3.5/4.5-12
3.5/4.5-13
3.5/4.5-14
3.5/4.5-15
3.5/4.5-16
3.5/4.5-17
3.6/4.6-9
3.6/4.6-10
3.6/4.6-30
3.6/4.6-31
3.6/4.6-32
3.6/4.6-33

INSERT

3.2/4.2-14
3.2/4.2-15
3.2/4.2-23*
3.2/4.2-24
3.5/4.5-7
3.5/4.5-8*
3.5/4.5-12*
3.5/4.5-13
3.5/4.5-14
3.5/4.5-15*
3.5/4.5-16
3.5/4.5-17*
3.6/4.6-9*
3.6/4.6-10
3.6/4.6-30*
3.6/4.6-31
3.6/4.6-32*
3.6/4.6-33*

*Denotes overleaf or spillover page.

9011090356 901030
PDR ADCK 05000259
P PNU

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	$\geq 470''$ above vessel zero	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level	$\geq 470''$ above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	$\geq 378''$ above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	$\geq 378''$ above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	$\geq 544''$ above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 and 62, SW #1)	$\geq 312 \frac{5}{16}''$ above vessel zero. A (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

BFN
Unit 1

3.2/4.2-14

TABLE 3.2.B (Continued)

BPN Unit 1	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
	2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
	2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2.5 psig	A	1. Above trip setting in con- junction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
	2(18)	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
	2(16)(18)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, drywell high pressure, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.

3.2/4.2-15



•
•
•
•

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

Action:

- A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.
 - B. Declare the system or component inoperable.
 - C. Immediately take action B until power is verified on the trip system.
 - D. No action required; indicators are considered redundant.
2. In only one trip system.
 3. Not considered in a trip system.
 4. Requires one channel from each physical location (there are 4 locations) in the steam line space.
 5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec. later.
 6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec. with similar pumps starting after about 14 sec. and 21 sec., at which time the full complement of CSS and RHRS pumps would be operating.
 7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.
 8. Note 1 does not apply to this item.
 9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.
11. In only two of the four 4160-V shutdown boards. See note 13.
12. In only one of the four 4160-V shutdown boards. See note 13.
13. An emergency 4160-V shutdown board is considered a trip system.
14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.
15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level (≥ 378 " above vessel zero) originating in the core spray system trip system.
16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.
17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.
18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.



1
2
3
4

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the **COLD SHUTDOWN CONDITION** within 24 hours.
9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be **OPERABLE**. The pumps' associated diesel generators must also be **OPERABLE**.
10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.
11. When there is irradiated fuel in the reactor and the reactor is not in the **COLD SHUTDOWN CONDITION**, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be **OPERABLE** and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

8. No additional surveillance required.
9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be **OPERABLE** shall be demonstrated to be **OPERABLE** per Specification 1.0.MM.
10. No additional surveillance required.
11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be **OPERABLE** per Specification 1.0.MM when the cross-connect capability is required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. If one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.
13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.
14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. No additional surveillance required.
13. No additional surveillance required.
14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.C RHR Service Water and Emergency
Equipment Cooling Water Systems
(EECWS) (Continued)

4. One of the D1 or D2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be inoperable for a period not to exceed 30 days provided the OPERABLE pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are OPERABLE.
5. The standby coolant supply capability may be inoperable for a period not to exceed 10 days.
6. If Specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the COLD SHUTDOWN CONDITION within 24 hours.
7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.

4.5.C RHR Service Water and Emergency
Equipment Cooling Water Systems
(EECWS) (Continued)

4. No additional surveillance is required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

E. High Pressure Coolant Injection System (HPCIS)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

E. High Pressure Coolant Injection System (HPCIS)

1. HPCI Subsystem testing shall be performed as follows:
 - a. Simulated Automatic Actuation Test Once/18 months
 - b. Pump OPERABILITY Per Specification 1.0.MM
 - c. Motor Operated Valve OPERABILITY Per Specification 1.0.MM
 - d. Flow Rate at normal reactor vessel operating pressure Once/3 months



•
•
•
•

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are OPERABLE.
3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

- e. Flow Rate at Once/18
150 psig months

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

- f. Verify that Once/Month
each valve
(manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCICS)

1. RCIC Subsystem testing shall be performed as follows:
 - a. Simulated Auto- Once/18
matic Actuation months
Test



•
•
•
•

LIMITING CONDITIONS FOR OPERATION

3.5.F Reactor Core Isolation Cooling System (RCICS)

3.5.F.1 (Cont'd)

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.
3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.F Reactor Core Isolation Cooling System (RCICS)

4.5.F.1 (Cont'd)

b. Pump	Per
OPERABILITY	Specifi-
	cation
	1.0.MM

c. Motor-Operated	Per
Valve	Specifi-
OPERABILITY	cation
	1.0.MM

d. Flow Rate at	Once/3
normal reactor	months
vessel operating	
pressure	

e. Flow Rate at	Once/18
150 psig	months

The RCIC pump shall deliver at least 600 gpm during each flow test.

f. Verify that	Once/Month
each valve	
(manual, power-	
operated, or	
automatic) in the	
injection flowpath	
that is not locked,	
sealed, or other-	
wise secured in	
position, is in its	
correct* position.	

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be OPERABLE:
 - (1) PRIOR TO STARTUP from a COLD CONDITION, or,
 - (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.
2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is OPERABLE. (Note that the pressure relief function of these valves is assured by Section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a HOT SHUTDOWN CONDITION in 6 hours, and in a COLD SHUTDOWN CONDITION in the following 18 hours.
3. If Specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be

4.5.G Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:
 - a. A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. Manual surveillance of the relief valves is covered in 4.6.D.2.
2. No additional surveillances are required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

3.5.G.3 (Cont'd)

initiated and the reactor vessel pressure shall be reduced to 105 psig or less within 24 hours.

H. Maintenance of Filled Discharge Pipe

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

4.5.G Automatic Depressurization System (ADS)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.
- b. Anytime the reactor is in RUN MODE, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN MODE except as defined in 3.6.C.1.c below.
- c. During the first 24 hours in the RUN MODE following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

4.6.C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.
2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3.6/4.6 BASES

3.6.C/4.6.C (Cont'd)

suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The two gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow at a reference pressure of (1,105 + 1 percent) psig. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6/4.6 BASES

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.
3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

3.6/4.6 BASES

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

3.6/4.6 BASES

3.6.G/4.6.G (Cont'd)

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)
2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code
3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)
4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)
5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)
6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)
7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

UNIT 2
EFFECTIVE PAGE LIST

REMOVE

3.2/4.2-14
3.2/4.2-15
3.2/4.2-16
3.2/4.2-17
3.2/4.2-23
3.2/4.2-24
3.5/4.5-7
3.5/4.5-8
3.5/4.5-12
3.5/4.5-13
3.5/4.5-14
3.5/4.5-15
3.5/4.5-16
3.5/4.5-17
3.6/4.6-9
3.6/4.6-10
3.6/4.6-30
3.6/4.6-31
3.6/4.6-32
3.6/4.6-33

INSERT

3.2/4.2-14
3.2/4.2-15
3.2/4.2-16*
3.2/4.2-17
3.2/4.2-23*
3.2/4.2-24
3.5/4.5-7
3.5/4.5-8*
3.5/4.5-12*
3.5/4.5-13
3.5/4.5-14
3.5/4.5-15*
3.5/4.5-16
3.5/4.5-17*
3.6/4.6-9*
3.6/4.6-10
3.6/4.6-30*
3.6/4.6-31
3.6/4.6-32*
3.6/4.6-33*

*Denotes overleaf or spillover page.

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq 470''$ above vessel zero.	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq 470''$ above vessel zero.	A	1. Multiplier relays initiate RCIC.
* 2	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	$\geq 378''$ above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	$\geq 378''$ above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, 105 sec. delay timer and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, 105 sec. delay timer, 12 1/2 min. delay timer, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	$\geq 544''$ above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A)	$\geq 312 \frac{5}{16}''$ above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

* The automatic initiation capability of this instrument channel is not required to be OPERABLE while the Reactor Vessel water level monitoring modification is being performed. Manual initiation capability of the associated system will be available during that time the automatic initiation logic is out-of-service.

BFN
Unit 2

3.2/4.2-14

TABLE 3.2.B (Continued)

BFN Unit 2	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
	2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
	2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 A-D)	≤ 2.5 psig	A	1. Above trip setting in con- junction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
3.2/4.2-15	2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
	2(16)(18)	Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, 105 sec. delay timer and CSS or RHR pump running, initiates ADS.

TABLE 3.2.B (Continued)

BFB Unit 2	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
	2	Instrument Channel - Reactor Low Pressure (PIS-3-74 A & B) (PIS-68-95, 96)	450 psig \pm 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
	2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B) (PS-68-95, 96)	230 psig \pm 15	A	1. Recirculation discharge valve actuation.
	1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig \pm 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
	2	Core Spray Auto Sequencing Timers (5)	6 \leq t \leq 8 sec.	B	1. With diesel power 2. One per motor
	2	LPCI Auto Sequencing Timers (5)	0 \leq t \leq 1 sec.	B	1. With diesel power 2. One per motor
	1	RHR SW A1, B3, C1, and D3 Timers	13 \leq t \leq 15 sec.	A	1. With diesel power 2. One per pump
	2	Core Spray and LPCI Auto Sequencing Timers (6)	0 \leq t \leq 1 sec. 6 \leq t \leq 8 sec. 12 \leq t \leq 16 sec. 18 \leq t \leq 24 sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
	1	RHR SW A1, B3, C1, and D3 Timers	27 \leq t \leq 29 sec.	A	1. With normal power 2. One per pump

3.2/4.2-16

TABLE 3.2.B (Continued)

BFN Unit 2	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
	1(16)	ADS Timer	105 sec \pm 7	A	1. Above trip setting in conjunction with low reactor water level permissive, low reactor water level, high drywell pressure or high drywell pressure bypass timer timed out, and RHR or CSS pumps running, initiates ADS.
	1(16)	ADS Timer (12 1/2 min.) (High Drywell Pressure Bypass Timer)	12 1/2 min. \pm 2	A	1. Above trip setting, in conjunction with low reactor water level permissive, low reactor water level, 105 sec. delay timer, and RHR or CSS pumps running, initiates ADS.
	2	Instrument Channel - RHR Discharge Pressure	100 \pm 10 psig	A	1. Below trip setting defers ADS actuation.
	2	Instrument Channel CSS Pump Discharge Pressure	185 \pm 10 psig	A	1. Below trip setting defers ADS actuation.
	1(3)	Core Spray Sparger to Reactor Pressure Vessel d/p	2 psid \pm 0.4	A	1. Alarm to detect core sparger pipe break.
	1	RHR (LPCI) Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
	1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
	1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.

3.2/4.2-17

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

Action:

- A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.
 - B. Declare the system or component inoperable.
 - C. Immediately take action B until power is verified on the trip system.
 - D. No action required; indicators are considered redundant.
2. In only one trip system.
 3. Not considered in a trip system.
 4. Requires one channel from each physical location (there are 4 locations) in the steam line space.
 5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec. later.
 6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec. with similar pumps starting after about 14 sec. and 21 sec., at which time the full complement of CSS and RHRS pumps would be operating.
 7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.
 8. Note 1 does not apply to this item.
 9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.
11. In only two of the four 4160-V shutdown boards. See note 13.
12. In only one of the four 4160-V shutdown boards. See note 13.
13. An emergency 4160-V shutdown board is considered a trip system.
14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.
15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level (\geq 378" above vessel zero) originating in the core spray system trip system.
16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.
17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.
18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.
9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE.
10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.
11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

8. No additional surveillance required.
9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.
10. No additional surveillance required.
11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. If three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.
13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.
14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. No additional surveillance required.
13. No additional surveillance required.
14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

4. Three of the D1, D2, B1, B2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be inoperable for a period not to exceed 30 days provided the OPERABLE pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are OPERABLE.
5. The standby coolant supply capability may be inoperable for a period not to exceed 10 days.
6. If Specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the COLD SHUTDOWN CONDITION within 24 hours.
7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.

4. No additional surveillance is required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

E. High Pressure Coolant Injection System (HPCIS)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

E. High Pressure Coolant Injection System (HPCIS)

1. HPCI Subsystem testing shall be performed as follows:
 - a. Simulated Automatic Actuation Test Once/18 months
 - b. Pump OPERABILITY Per Specification 1.0.MM
 - c. Motor Operated Valve OPERABILITY Per Specification 1.0.MM
 - d. Flow Rate at normal reactor vessel operating pressure Once/3 months

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

- e. Flow Rate at Once/18
150 psig months

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS(LPCI), and RCIGS are OPERABLE.

2. No additional surveillances are required.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

- * Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCIGS)

F. Reactor Core Isolation Cooling System (RCIGS)

1. The RCIGS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows:

- a. Simulated Auto- Once/18
matic Actuation months
Test

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.F. Reactor Core Isolation Cooling System (RCICS)

4.5.F Reactor Core Isolation Cooling System (RCICS)

3.5.F.1 (Cont'd)

4.5.F.1 (Cont'd)

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

b. Pump OPERABILITY Per Specification 1.0.MM

c. Motor-Operated Valve OPERABILITY Per Specification 1.0.MM

d. Flow Rate at normal reactor vessel operating pressure Once/3 months

e. Flow Rate at 150 psig Once/18 months

The RCIC pump shall deliver at least 600 gpm during each flow test.

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be OPERABLE:
 - (1) PRIOR TO STARTUP from a COLD CONDITION, or,
 - (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUT-DOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.
2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is OPERABLE. (Note that the pressure relief function of these valves is assured by Section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a HOT SHUTDOWN CONDITION in 6 hours, and in a COLD SHUTDOWN CONDITION in the following 18 hours.
3. If Specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be initiated and the reactor

4.5.G Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:
 - a. A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. Manual surveillance of the relief valves is covered in 4.6.D.2.
2. No additional surveillances are required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

3.5.G.3 (Cont'd)

vessel pressure shall be reduced to 105 psig or less within 24 hours.

H. Maintenance of Filled Discharge Pipe

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

Pl-75-20	48 psig
Pl-75-48	48 psig
Pl-74-51	48 psig
Pl-74-65	48 psig

4.5.G Automatic Depressurization System (ADS)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.



• •
• •
• •
• •

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.
- b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.
- c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

4.6.C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.
2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3.6/4.6 BASES

3.6.B/4.6.C (Cont'd)

five gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The 2 gpm limit for coolant leakage rate increases over any 24-hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6/4.6 BASES

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.
3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety--Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

3.6/4.6 BASES

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Operation without forced recirculation is permitted for up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be operable while in the RUN mode provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)
2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code
3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)
4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)
5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)
6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)
7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

UNIT 3
EFFECTIVE PAGE LIST

REMOVE

3.2/4.2-14
3.2/4.2-15
3.2/4.2-22
3.2/4.2-23
3.5/4.5-7
3.5/4.5-8
3.5/4.5-12
3.5/4.5-13
3.5/4.5-14
3.5/4.5-15
3.5/4.5-16
3.5/4.5-17
3.6/4.6-9
3.6/4.6-10
3.6/4.6-30
3.6/4.6-31
3.6/4.6-32
3.6/4.6-33

INSERT

3.2/4.2-14
3.2/4.2-15
3.2/4.2-22*
3.2/4.2-23
3.5/4.5-7
3.5/4.5-8*
3.5/4.5-12*
3.5/4.5-13
3.5/4.5-14
3.5/4.5-15*
3.5/4.5-16
3.5/4.5-17*
3.6/4.6-9*
3.6/4.6-10
3.6/4.6-30*
3.6/4.6-31
3.6/4.6-32*
3.6/4.6-33*

*Denotes overleaf or spillover page.

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

BFN
Unit 3

Minimum No.
Operable Per
Trip Sys(1)

3.2/4.2-14

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	$\geq 470''$ above vessel zero.	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level	$\geq 470''$ above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW#1)	$\geq 378''$ above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW#2)	$\geq 378''$ above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW#1)	$\geq 544''$ above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 and 62, SW#1)	$\geq 312 \frac{5}{16}''$ above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

TABLE 3.2.B (Continued)

BFN Unit 3	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
	2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
	2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW#2)	≤ 2.5 psig	A	1. Above trip setting in con- junction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
	2(18)	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW#1)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
3.2/4.2-15	2(16)(18)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, drywell high pressure, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

Action:

- A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.
 - B. Declare the system or component inoperable.
 - C. Immediately take action B until power is verified on the trip system.
 - D. No action required; indicators are considered redundant.
2. In only one trip system.
 3. Not considered in a trip system.
 4. Requires one channel from each physical location (there are 4 locations) in the steam line space.
 5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 seconds later.
 6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 seconds with similar pumps starting after about 14 seconds and 21 seconds, at which time the full complement of CSS and RHRS pumps would be operating.
 7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.
 8. Note 1 does not apply to this item.
 9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.



•
•
•
•

NOTES FOR TABLE 3.2.B (Continued)

10. Only one trip system for each cooler fan.
11. In only two of the four 4160-V shutdown boards. See note 13.
12. In only one of the four 4160-V shutdown boards. See note 13.
13. An emergency 4160-V shutdown board is considered a trip system.
14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.
15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level (≥ 378 " above vessel zero) originating in the core spray system trip system.
16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.
17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.
18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.
9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE.
10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.
11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

8. No additional surveillance required.
9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.
10. No additional surveillance required.
11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. If one RHR pump or associated heat exchanger located on the unit cross-connection in unit 2 is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.
13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.
14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

12. No additional surveillance required.
13. No additional surveillance required.
14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

4. One of the B1 or B2 RHRSW pumps assigned to the RHR heat exchanger supplying the standby coolant supply connection may be inoperable for a period not to exceed 30 days provided the OPERABLE pump is aligned to supply the RHR heat exchanger header and the associated diesel generator and essential control valves are OPERABLE.
5. The standby coolant supply capability may be inoperable for a period not to exceed 10 days.
6. If Specifications 3.5.C.2 through 3.5.C.5 are not met, an orderly shutdown shall be initiated and the unit placed in the COLD SHUTDOWN CONDITION within 24 hours.
7. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel.

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

4. No additional surveillance is required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

E. High Pressure Coolant Injection System (HPCIS)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

E. High Pressure Coolant Injection System (HPCIS)

1. HPCI Subsystem testing shall be performed as follows:

a. Simulated Automatic Actuation Test	Once/18 months
b. Pump OPERABILITY	Per Specification 1.0.MM
c. Motor Operated Valve OPERABILITY	Per Specification 1.0.MM
d. Flow Rate at normal reactor vessel operating pressure	Once/3 months

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

- e. Flow Rate at Once/18
150 psig months

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are OPERABLE.

2. No additional surveillances are required.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

- * Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCICS)

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows:
a. Simulated Auto- Once/18
matic Actuation months
Test

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.F Reactor Core Isolation Cooling System (RCICS)

4.5.F Reactor Core Isolation Cooling System (RCICS)

3.5.F.1 (Cont'd)

4.5.F.1 (Cont'd)

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

b. Pump OPERABILITY Per Specification 1.0.MM

c. Motor-Operated Valve OPERABILITY Per Specification 1.0.MM

d. Flow Rate at normal reactor vessel operating pressure Once/3 months

e. Flow Rate at 150 psig Once/18 months

The RCIC pump shall deliver at least 600 gpm during each flow test.

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be OPERABLE:
 - (1) PRIOR TO STARTUP from a COLD CONDITION, or,
 - (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUT-DOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.
2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is OPERABLE. (Note that the pressure relief function of these valves is assured by Section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a HOT SHUTDOWN CONDITION in 6 hours, and in a COLD SHUTDOWN CONDITION in the following 18 hours.
3. If Specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be initiated and the reactor

4.5.G Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:
 - a. A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. Manual surveillance of the relief valves is covered in 4.6.D.2.
2. No additional surveillances are required.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

3.5.G.3 (Cont'd)

vessel pressure shall be reduced to 105 psig or less within 24 hours.

H. Maintenance of Filled Discharge Pipe

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

4.5.G Automatic Depressurization System (ADS)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

3.6.C Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.
- b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.
- c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

4.6.C Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

D. Relief Valves

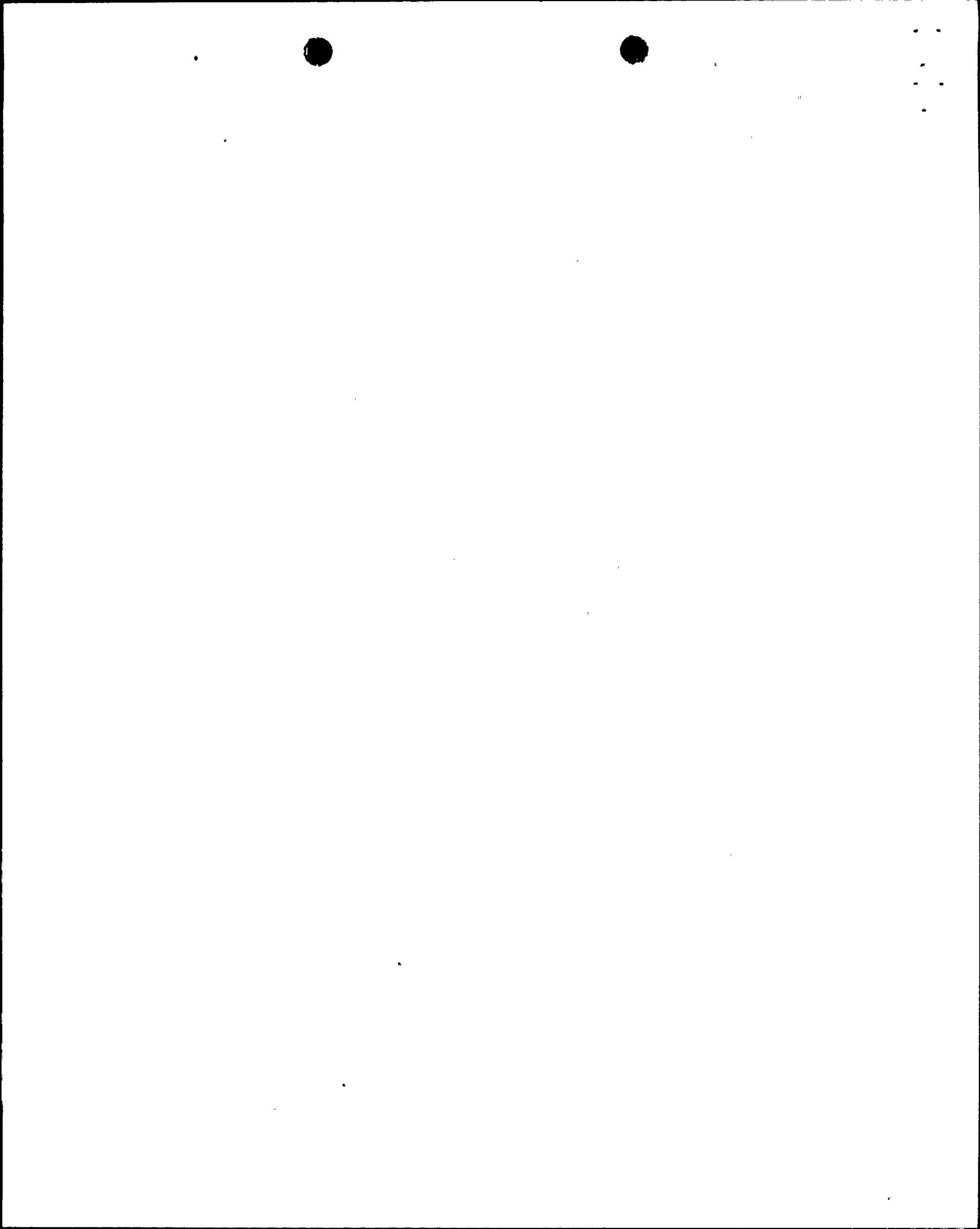
1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.
2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.



3.6/4.6 BASES

3.6.C/4.6.C (Cont'd)

suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The two gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

References

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 83.77 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief and safety valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6/4.6 BASES

3.6.D/4.6.D (Cont'd)

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

References

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
3. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow

3.6/4.6 BASES

3.6.E/4.6.E (Cont'd)

area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

3.6/4.6 BASES

3.6.G/4.6.G (Cont'd)

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

References

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)
2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code
3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)
4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

Enclosure 2

REASON FOR CHANGES, DESCRIPTION, AND JUSTIFICATION
BROWNS FERRY NUCLEAR PLANT (BFN)

REASON FOR CHANGES

BFN units 1, 2, and 3 technical specifications (TSs) are being changed to: (1) revise Table 3.2.B and Limiting Conditions for Operation (LCO) 3.5.B.11, 3.5.E.1, 3.5.F.1, 3.5.G.1, and 3.6.D.1 and the bases section for 3.6.D/4.6.D to clarify equipment operability requirements when the reactor is in the cold shutdown condition, (2) revise the maximum operating power level allowed with an inoperable Recirculation Pump Trip (RPT) system(s) from 85 percent to 30 percent power, and (3) correct two typographical errors in Table 3.2.B.

DESCRIPTION AND JUSTIFICATION FOR THE PROPOSED CHANGES

1. Clarify equipment operability requirements for when the reactor is in the cold shutdown condition.
 - a. Add the following note to Table 3.2.B for the drywell high pressure instruments (PIS-64-57 A-D, PIS-64-58 A-D, and PIS-64-58 E-H for unit 2 and PS-64-57 A-D, PS-64-58 A-D, and PS-64-58 E-H for units 1 and 3):

"18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION."
 - b. Existing LCO 3.5.B.11 reads in part: "When there is irradiated fuel in the reactor and the reactor vessel pressure is greater than atmospheric, . . ."

Revised LCO 3.5.B.11 would read in part: "When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, . . ."
 - c. Existing LCO 3.5.E.1 reads in part: "The HPCI system shall be OPERABLE . . . except as specified in Specification 3.5.E.2 . . ."

Revised LCO 3.5.E.1 would read in part: "The HPCI system shall be OPERABLE . . . except in the COLD SHUTDOWN CONDITION or as specified in 3.5.E.2 . . ."
 - d. Existing LCO 3.5.F.1 reads in part: "The RCICS shall be OPERABLE . . . except as specified in 3.5.F.2. . ."

Revised LCO 3.5.F.1 would read in part: "The RCICS shall be operable . . . except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2 . . ."
 - e. Existing LCO 3.5.G.1 reads in part: "Four of the six valves of the Automatic Depressurization System shall be OPERABLE . . . except as specified in 3.5.G.2 and 3.5.G.3 below."

Revised LCO 3.5.G.1 would read in part: "Four of the six valves of the Automatic Depressurization System shall be OPERABLE . . . except in the COLD SHUTDOWN CONDITION or as specified in 3.5.G.2 or 3.5.G.3 below."



11
12
13
14

- f. Existing LCO 3.6.D.1 reads in part: "When more than one relief valves are known to be failed . . ."

Revised LCO 3.6.D.1 would read in part: "When more than one relief valve is known to be failed . . ."

- g. Add the following to LCO 3.6.D.1: "The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION."
- h. Add the following paragraph to the bases section for 3.6.D/4.6.D:
"The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy."

Table 3.2.B currently requires the drywell high pressure instruments to be operable whenever any core and containment cooling system in TS section 3.5 is operable. Change (a) above would allow these instruments to be inoperable when the plant is in the cold shutdown condition. Initiation of these instruments, along with the low reactor pressure instruments, indicates a breach of the nuclear system process barrier within the drywell (steam leak). With the reactor in the cold shutdown condition (reactor coolant temperature $\leq 212^{\circ}\text{F}$ and reactor in shutdown or refuel mode), there is no need to detect steam leaks so it is acceptable for the drywell high pressure instruments to be inoperable.

Additionally, TVA is required by TS section 4.7.A to conduct an Integrated Leakrate Test (ILRT) at certain frequencies. The ILRT demonstrates the ability of the primary containment to contain the gases released from the primary system during a postulated worst case accident with leakage rates low enough to ensure exposure rates less than the 10 CFR 100 limits. The test is performed with the plant in the cold shutdown condition by pressurizing the primary containment (drywell and torus) to design bases accident pressure (49.6 psig) and monitoring pressure and temperature for a prescribed period of time. From this data, the leakage can be calculated.

The high drywell pressure instruments listed above have a trip level setting of between 1 and 2.5 psig. Inhibiting these pressure instruments during the ILRT is required to prevent unnecessary Emergency Core Cooling System (ECCS) initiations.

The Residual Heat Removal (RHR) and Core Spray (CS) systems are required to be operable during the test in accordance with LCOs 3.5.A and 3.5.B because reactor pressure is greater than atmospheric. The reactor low water level instruments (LS-3-58 A-D) are operable and initiate the RHR or CS systems on a low-low reactor water level if necessary. This ensures that RHR and CS could provide sufficient makeup capacity if required to protect the fuel.



11
12
13

The standby gas treatment and secondary containment systems are also operable during the test and available to contain and filter any radioactive material were it to be released.

TSs currently require that the RHR crosstie be operable with reactor pressure greater than atmospheric (LCO 3.5.B.11), the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems must be operable with reactor pressure greater than 150 psig, and the Automatic Depressurization System (ADS) and relief valves must be operable with reactor pressure greater than 105 psig.

With the reactor in the cold shutdown condition, vessel pressure is atmospheric except during performance of inservice hydrostatic pressure tests, inservice leakage tests and ILRT.

Once per inservice inspection interval, the plant is required to perform an inservice hydrostatic pressure test on the reactor vessel and attached piping out to and including the first isolation valve to ensure the retaining capability of the reactor coolant pressure boundary. The test is performed at pressures (1096 to 1150 psia in the dome) in excess of normal operating pressure (approximately 1020 psia).

An inservice leakage test is required whenever the reactor coolant pressure retaining boundary is breached. This test is similar to the hydrostatic test, but it is performed at normal operating pressure (approximately 1020 psia). As described previously, the ILRT is performed by pressurizing the primary containment to 49.6 psig.

As currently written, TSs require the RHR crosstie to be operable for each of these tests and HPCI, RCIC, ADS, and relief valves to be operable for the inservice hydrostatic and leakage tests. In reality, the ADS is disabled and HPCI and RCIC, both steam turbine driven systems, have no steam supply available during these tests.

These tests are performed in the cold shutdown condition at the end of the refueling outage with fuel loaded and the reactor pressure vessel head installed. These tests occur when primary system energy is minimal with all control rods inserted. Because the reactor vessel pressure is greater than atmospheric the RHR and CS systems are required to be operable.

The inservice hydrostatic and leakage tests are performed at or above a minimum temperature as specified by TS figure 3.6-1. With the system temperature (approximately 207°F) below the atmospheric pressure boiling point, enthalpy of the bulk fluid is low. If a leak greater than the makeup capacity of the Control Rod Drive (CRD) pump should occur during the test, the system would be depressurized well below the maximum pressure at which the RHR and CS systems could inject to the vessel before water level dropped to an unsafe level. The available RHR and CS systems are sufficient to preclude fuel uncovering in the event of a leak.



11
12
13

1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48 49 50 51 52 53 54 55 56 57 58 59 60 61 62 63 64 65 66 67 68 69 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 100

1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41
42
43
44
45
46
47
48
49
50
51
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
94
95
96
97
98
99
100

The RHR crosstie is provided to maintain a long term reactor core and primary containment cooling capability irrespective of primary containment integrity or operability of the RHR system on a given unit. This is provided in case the torus is breached, flooding the RHR pumps on the affected unit. However, with the reactor in the cold shutdown condition, there is no high energy potential to breach the torus so the RHR crosstie is not needed.

During inservice hydrostatic testing, 11 of the 13 relief valves are disabled by removing the pilot cartridges and blanking the pilot ports. Overpressure protection is provided by the two remaining relief valves which have their setpoint established in accordance with ASME Section XI. The relief capacity of one relief valve exceeds the flow capacity of the hydrostatic pressure source. Two valves are used for redundancy.

These changes are consistent with the General Electric Boiling Water Reactor Standard TSs (NUREG 0123) which requires HPCI (section 3.5.1.c), RCIC (section 3.7.4), ADS (section 3.5.1.d), and relief valves (section 3.4.2.1) to be operable only in the power operation, startup, and hot shutdown conditions.

2. Correct the maximum power level allowed with an inoperable RPT system(s).

Existing Table 3.2.B, Note 17 reads:

"17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if 1 RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 85 percent within four hours."

Proposed change to Table 3.2.B, Note 17 would read:

"17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours."

This change corrects the maximum operating power level allowed with an inoperable RPT system(s) from 85 percent to 30 percent Core Thermal Power (CTP). Thirty percent CTP is used in the RPT analysis (NEDO-24119, "Basis for Installation of Recirculation Pump Trip System for Browns Ferry," April 1978, BFN Updated Final Safety Analysis Report (UFSAR) Section 7.9.4.5), and is conservatively determined to be the maximum power level at which fuel cladding integrity can be assumed during an end of cycle limiting overpressurization event without RPT protection.

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

100-100000

The RPT provides automatic trip of both recirculation pumps after a turbine trip or generator load rejection if reactor power is above approximately 30 percent of rated full load. The purpose of this trip is to reduce the peak reactor pressure and peak heat flux resulting from transients in which it is postulated that there is a coincident failure of the turbine bypass system. The recirculation pump trip signal results from either turbine control valve fast closure or turbine stop valve closure. Reactor scram is also initiated by these signals. The very rapid reduction in core flow following a recirculation pump trip early in these transients reduces the severity of these events because the immediate resultant increase in core voids provides negative reactivity which supplements the negative reactivity from control rod scram.

The proposed change reduces the trip set point from 85 percent to 30 percent and is therefore more conservative than the current operational requirements.

Additionally, the number "1" in Note 17 is being revised to the alphabetic "one" to be consistent with the rest of the note.

3. Correct two typographical errors in Table 3.2.B.

- a. Correct typographical error in the first entry under "Remarks" in Table 3.2.B (Page 3.2/4.2-14).

Existing entry reads:

"1. Below trip setting initiated HPCI."

Proposed change to Table 3.2.B would read:

"1. Below trip setting initiates HPCI."

This change revises the word "initiated" to "initiates" so that this entry will be in the present tense like the other remarks in Table 3.2.B.

- b. Correct typographical error under "Minimum No. Operable Per Trip Sys" column on Table 3.2.B for "RHR (LPCI) Trip System bus power monitor" (page 3.2/4.2-17) Unit 2 only.

The entry in this column should be "1."

This is an omission in the Unit 2 TSs. The original BFN Unit 2 TSs indicate a "1" in this column as do the current BFN Units 1 and 3 TSs.

PROPOSED DETERMINATION OF NO SIGNIFICANT HAZARDS
BROWNS FERRY NUCLEAR PLANT (BFN)

DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATION (TS) AMENDMENT

BFN units 1, 2, and 3 technical specifications (TSs) are being changed to: (1) revise Table 3.2.B and Limiting Conditions for Operation (LCO) 3.5.B.11, 3.5.E.1, 3.5.F.1, 3.5.G.1, and 3.6.D.1 and the bases section for 3.6.D/4.6.D to clarify equipment operability requirements when the reactor is in the cold shutdown condition, (2) revise the maximum operating power level allowed with an inoperable Recirculation Pump Trip (RPT) system(s) from 85 percent to 30 percent power, and (3) correct two typographical errors in Table 3.2.B.

BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from an accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. Change 1 clarifies equipment operability requirements with the reactor in the cold shutdown condition. With the reactor in the cold shutdown condition, primary system energy is minimal and the control rods are inserted. Reactor pressure is normally atmospheric except during performance of inservice hydrostatic tests, inservice leakage tests, and Integrated Leak Rate Tests (ILRT). This change would inhibit the drywell high pressure instruments which function to detect primary system leaks. With minimal system energy and no steam generation, this function is not required. The High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems are not required because there is no steam supply to operate them and Residual Heat Removal (RHR) and Core Spray (CS) are operable and capable of providing makeup in case of leaks to protect the fuel from being uncovered. The Automatic Depressurization System (ADS) is not required for leaks considered possible during the inservice hydrostatic test. Reactor pressure would decrease fast enough to allow residual heat removal and core spray injection in time to preclude water level decreasing to an unsafe level. The relief valves are not required to be operable because alternate means of overpressurization protection are provided in the tests. During inservice hydrostatic testing, 11 of the 13 relief valves are disabled by removing the pilot cartridges and blanking the pilot ports. Overpressure protection is provided by the two remaining relief valves which have their setpoint established in accordance with ASME Section XI. The RHR crosstie is not required because there is no high energy potential to breach the torus in the cold shutdown condition. The change is consistent with industry practice and the GE BWR Standard TSs (NUREG 0123).



100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

Change 2 is a more conservative requirement. The RPT system provides an automatic trip of both recirculation pumps after a turbine trip or a generator load reject. This reduction in flow increases the core voids and provides immediate negative reactivity to reduce the severity of the transient. There are two RPT systems. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, reactor power shall be less than 30 percent within four hours (vs. the current 85 percent). The proposed value of 30 percent power is consistent with the BFN RPT analysis and the BFN Updated Final Safety Analysis Report. Therefore, this change involves no significant increase in the probability or consequences of an accident previously analyzed.

Change 3 is an administrative change that corrects typographical errors.

2. The proposed change does not create the possibility of a new or different kind of accident from an accident previously evaluated. Change 1 does not involve changes in plant hardware or method of operation from that currently practiced. The changes are clarifications to TSs to facilitate performance of required TS testing with the reactor in the cold shutdown condition. The methods of performance are consistent with industry practice.

Change 2 will ensure that when both RPT systems are inoperable or when one RPT system is inoperable more than 72 hours, reactor power is dropped to a level consistent with the analysis performed for the RPT installation.

Change 3 corrects two typographical errors so the TSs will be more consistent.

3. The proposed changes do not involve a significant reduction in the margin of safety. Change 1 clarifies equipment operability requirements with the reactor in the cold shutdown condition. Sufficient safety equipment is still available to ensure the fuel remains covered, even in the event of leaks. It does not reduce the equipment available to mitigate an accident and as such does not reduce the margin of safety.

Change 2 is more conservative than the current TS. When the RPT system is inoperable the maximum allowed reactor power will be reduced. This is consistent with the analysis performed for the RPT installation and the FSAR and does not reduce the margin of safety.

Change 3 is an administrative change which does not reduce the margin of safety.

These changes have been reviewed by TVA. Based on this review, TVA does not believe the changes present a Significant Hazards Consideration.

云
山
行
記