ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATIONS REVISIONS

BROWNS FERRY NUCLEAR PLANT

UNITS 1, 2, AND 3

(TVA BFN TS 249)

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-58A LI-3-58B	Reactor Water Level	Indicator - 155" to +60"	(1) (2) (3)
1	LI-3-46A	Reactor Water Level	Indicator - 155" to 60"	(9)
2	PI-3-74A PI-3-74B	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
1	PI-3-79	Reactor Pressure	Indicator 0-1200 psig	(9)
2	XR-64-50 PI-64-67B TI-64-52AB	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	XR-64-50	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	XR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights)	
1	N/A	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	(1) (2) (3) (4)
1	PS-64-67B	Drywell Pressure	Alarm at 35 psig)	
1	TS-64-52A & PIS-64-58A & IS-64-67A	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure >2.5 psig after 30 minute delay)	(1) (2) (3) (4)
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding 30 days unless such instrumentation is sooner made OPERABLE.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made OPERABLE.

(3) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) From and after the date that both the acoustic monitor and the temperature indication on any one valve fails to indicate in the control room, continued operation is permissible during the succeeding 30 days, unless one of the two monitoring channels is sooner made OPERABLE. If both the primary and secondary indication on any SRV tailpipe is INOPERABLE, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV.

(6) A channel consists of eight sensors, one from each alternating torus bay. Seven sensors must be OPERABLE for the channel to be OPERABLE.

(7) When one of these instruments is INOPERABLE for more than seven days, in lieu of any other report required by Specification 6.7.2, prepare and submit a Special Report to the Commission pursuant to Specification 6.7.3 within the next seven days outlining the action taken, the cause of inoperability, and the plans and schedule for restoring the system to OPERABLE status.

(8) With the plant in the power operation, Startup, or Hot Shutdown condition and with the number of OPERABLE channels less than the required OPERABLE channels, either restore the INOPERABLE channel(s) to OPERABLE status within 72 hours, or initiate the preplanned alternate method of monitoring the appropriate parameter.

(9) If this instrument is inoperable, establish within the next hour a patrolling fire watch in fire area 16 to ensure that the affected fire area is checked hourly.

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level (LI-3-46A,58A&B)	Once/6 months	Each Shift
2) Reactor Pressure (PI-3-79,74A&B)	Once/12 months	Each Shift
3) Drywell Pressure (PI-64-67B) and XR-64-50	Once/6 months	Each Shift
4) Drywell Temperature (TI-64-52AB) and XR-64-50	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature (XR-64-52)	Once/6 months	Each Shift
8) Control Rod Position	N/A	Each Shift
9) Neutron Monitoring	(2)	Each Shift
10) Drywell Pressure (PS-64-67B)	Once/6 months	N/A
11) Drywell Pressure (PIS-64-58A)	Once/6 months	N/A
12) Drywell Temperature (TS-64-52A)	Once/6 months	N/A
13) Timer (IS-64-67A)	Once/6 months	N/A
14) CAD Tank Level	Once/6 months	Once/day
15) Containment Atmosphere Monitors	Once/6 months	Once/day

Trip setting of 100 mr/hr for the monitors in the refueling zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

For each parameter monitored, as listed in Table 3.2.F, there are two channels of instrumentation except as noted. By comparing readings between the two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings. A single channel of instruments at the backup control panel provides the additional indication of reactor vessel water level and reactor pressure. This indication is available to ensure safe shutdown capability from outside the control room.

Instrumentation is provided for isolating the control room and initiating a pressurizing system that processes outside air before supplying it to the control room. An accident signal that isolates primary containment will also automatically isolate the control room and initiate the emergency pressurization system. In addition, there are radiation monitors in the normal ventilation system that will isolate the control room and initiate the emergency pressurization system. Activity required to cause automatic actuation is about one mRem/hr.

Because of the constant surveillance and control exercised by TVA over the Tennessee Valley, flood levels of large magnitudes can be predicted in advance of their actual occurrence. In all cases, full advantage will be taken of advance warning to take appropriate action whenever reservoir levels above normal pool are predicted; however, the plant flood protection is always in place and does not depend in any way on advanced warning. Therefore, during flood conditions, the plant will be permitted to operate until water begins to run across the top of the pumping station at elevation 565. Seismically qualified, redundant level switches each powered from a separate division of power are provided at the pumping station to give main control room indication of this condition. At that time an orderly shutdown of the plant will be initiated, although surges even to a depth of several feet over the pumping station deck will not cause the loss of the main condenser circulating water pumps.

TABLE 3.5-1

MINIMUM RHRSW AND EECW PUMP ASSIGNMENT

Time Limit (Days)	Minimum Service Assignment RHRSW	EECW(B)
Indefinite	(D)(E)(G) 7	(A)(H) 4
30	(C)(D)(E)(F)(G) 7 or 6	(A)(C)(F)(H) 2 or 3
7	(D)(E)(G) 6	(A)(H) 2

(A) At least one operable pump must be assigned to each header.

(B) Only automatically starting pumps may be assigned to EECW header service.

(C) Nine pumps must be OPERABLE. Either configuration is acceptable: 7 and 2 or 6 and 3 (except reduced by notes D and E).

(D) Requirements may be reduced by two for each unit with fuel unloaded.

(E) For units with fuel loaded, the minimum RHRSW pump requirements may be reduced by one pump for each unit that has been in COLD SHUTDOWN CONDITION for more than 96 hours. At least 2 of the required pumps must be powered from separate electric power sources with their associated RHR pumps, heat exchangers, and diesel generator(s) OPERABLE.

(F) These minimum service requirements are also applicable to startup from a COLD SHUTDOWN CONDITION.

(G) RHRSW pumps D2 and either C1 or C2 must be OPERABLE during unit 2 REACTOR POWER OPERATION. If D2 or both C1 and C2 pumps are inoperable, within the next hour establish a patrolling fire watch in fire areas/zones shown in Table 3.5-2 to ensure the affected fire areas/zones are checked hourly.

(H) EECW pumps A3, B3, C3, and D3 must be OPERABLE during unit 2 REACTOR POWER OPERATION. If one or more of these pumps is inoperable, within the next hour establish a patrolling fire watch in fire areas/zones shown in Table 3.5-2 to ensure the affected fire areas/zones are checked hourly.

TABLE 3.5-2

RHRSW/EECW Pump Inoperable	Fire Areas/Zones to Establish Patrolling Fire Watch
C1 and C2	2-2, 2-5, 16, 18
D2	2-1, 2-3, 2-4, 2-6, 9
A3	2-1, 2-2, 2-3, 2-4, 2-5, 2-6, 9
B3	16, 18
C3	2-1, 2-2, 2-3, 2-4, 2-5, 2-6, 9
D3	16, 18

3.5 Bases (Cont'd)

The suppression chamber can be drained when the reactor vessel pressure is atmospheric, irradiated fuel is in the reactor vessel, and work is not in progress which has the potential to drain the vessel. By requiring the fuel pool gate to be open with the vessel head removed, the combined water inventory in the fuel pool, the reactor cavity, and the separator/dryer pool, between the fuel pool low level alarm and the reactor vessel flange, is about 65,800 cubic feet (492,000 gallons). This will provide adequate low-pressure cooling in lieu of CSS and RHR (LPCI and containment cooling mode) as currently required in Specifications 3.5.A.4 and 3.5.B.9. The additional requirements for providing standby coolant supply available will ensure a redundant supply of coolant supply. Control rod drive maintenance may continue during this period provided no more than one drive is removed at a time unless blind flanges are installed during the period of time CRDs are not in place.

Should the capability for providing flow through the cross-connect lines be lost, a 10-day repair time is allowed before shutdown is required. This repair time is justified based on the very small probability for ever needing RHR pumps and heat exchangers to supply an adjacent unit.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR subsection 4.8)

2. Core Standby Cooling Systems (BFNP FSAR Section 6)

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

There are two EECW headers (north and south) with four automatic starting RHRSW pumps on each header. All components requiring emergency cooling water are fed from both headers thus assuring continuity of operation if either header is operable. Each header alone can handle the flows to all components. Two RHRSW pumps can supply the full flow requirements of all essential EECW loads for any abnormal or postaccident situation.

In fire areas 9, 16 and 18 and fire zones 2-1, 2-2, 2-3, 2-4, 2-5, and 2-6, a postulated fire could result in only two EECW pumps being available that are required by the plant Appendix R evaluation. If one of these two remaining EECW pumps was the one allowed by the technical specifications to be indefinitely out of service, then the required two EECW pumps for safe shutdown would not be available. If one of the required EECW pumps is out of service, a hourly patrolling fire watch will be established in the appropriate fire area/zones as a compensatory measure. For a fire in any other areas/zones of the plant, adequate RHRSW swing/EECW pumps are available to supply necessary cooling water to the diesel generators, even if one of the EECW pumps is out of service.

There are four RHR heat exchanger headers (A, B, C, & D) with one RHR heat exchanger from each unit on each header. There are two RHRSW pumps on each header; one normally assigned to each header (A2, B2, C2, or D2) and one on alternate assignment (A1, B1, C1, or D1). One RHR heat exchanger header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three RHRSW heat exchangers on the header. One RHRSW pump can supply the full flow requirement of one RHR heat exchanger. Two RHR exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation.

The RHR Service Water System was designed as a shared system for three units. The specification, as written, is conservative when consideration is given to particular pumps being out of service and to possible valving arrangements. If unusual operating conditions arise such that more pumps are out of service than allowed by this specification, a special case request may be made to the NRC to allow continued operation if the actual system cooling requirements can be assured.

Should three of the four RHRSW pumps normally or alternately assigned to the RHR heat exchanger headers supplying the standby coolant supply connection become inoperable, capability for long-term fluid makeup to the unit reactor and for cooling of the unit containment remains operable. Because of the availability of makeup and cooling capability which is demonstrated to be operable immediately and with specified subsequent surveillance, a 30-day repair period is justified. Unit 2 may be supplied standby coolant from either of four pumps--B1, B2, D1, or D2. Should the capability to provide standby coolant supply be lost, a 10-day repair time is justified based on the low probability for ever needing the standby coolant supply.

The plant Appendix R evaluation requires that either RHRSW pump C1 or D2 be available, however both pumps are required to be operable to ensure the one required RHRSW pump is available for a specific fire location. If one of the two required RHRSW pumps is out of service, a hourly patrolling fire watch will be established in the appropriate fire areas/zones as a compensatory measure.

3.5.D Equipment Area Coolers

There is an equipment area cooler for each RHR pump and an equipment area cooler for each set (two pumps, either the A and C or B and D pumps) of core spray pumps. The equipment area coolers take suction near the cooling air discharge of the motor of the pump(s) served and discharge air near the cooling air suction of the motor of the pump(s) served. This ensures that cool air is supplied for cooling the pump motors.

3.5 BASES (Cont'd)

The equipment area coolers also remove the pump, and equipment waste heat from the basement rooms housing the engineered safeguard equipment. The various conditions under which the operation of the equipment air coolers is required have been identified by evaluating the normal and abnormal operating transients and accidents over the full range of planned operations. The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. This testing is adequate to assure the operability of the equipment area coolers.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR paragraphs 4.8.9.1 and 4.8.9.2)

2. Core Standby Cooling System (BFNP FSAR subsection 6.7)

3.5.E. High Pressure Coolant Injection System (HPCIS)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCIS permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or core spray system operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 5,000 gpm at reactor pressures between 1,120 and 150 psig. Two sources of water are available. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI system. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization caused the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

The minimum required NPSH for HPCI is 21 feet. There is adequate elevation head between the suppression pool and the HPCI pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

3.5 BASES (Cont'd)

The HPCIS serves as a backup to the RCICS as a source of feedwater makeup during primary system isolation conditions. The ADS serves as a backup to the HPCIS for reactor depressurization for postulated transients and accident. Both these systems are checked for operability if the HPCI is determined to be inoperable. Considering the redundant systems, an allowable repair time of seven days was selected.

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be operable when starting up from a Cold Condition. It is realized that the HPCI is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before the reactor pressure decreases below 100 psig. It is the intent of this specification to assure that when the reactor is being started up from a Cold Condition, the HPCI is not known to be inoperable.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The various conditions under which the RCICS plays an essential role in providing makeup water to the reactor vessel have been identified by evaluating the various plant events over the full range of planned operations. The specifications ensure that the function for which the RCICS was designed will be available when needed. The minimum required NPSH for RCIC is 20 feet. There is adequate elevation head between the suppression pool and the RCIC pump, such that the required NPSH is available with a suppression pool temperature up to 140°F with no containment back pressure.

Because the low-pressure cooling systems (LPCI and core spray) are capable of providing all the cooling required for any plant event when nuclear system pressure is below 122 psig, the RCICS is not required below this pressure. Between 122 psig and 150 psig the RCICS need not provide its design flow, but reduced flow is required for certain events. RCICS design flow (600 gpm) is sufficient to maintain water level above the top of the active fuel for a complete loss of feedwater flow at design power (105 percent of rated).

Consideration of the availability of the RCICS reveals that the average risk associated with failure of the RCICS to cool the core when required is not increased if the RCICS is inoperable for no longer than seven days, provided that the HPCIS is operable during this period.

REFERENCE

1. Reactor Core Isolation Cooling System (BFNP FSAR Subsection 4.7)

3.5.G Automatic Depressurization System (ADS)

This specification ensures the operability of the ADS under all conditions for which the depressurization of the nuclear system is an essential response to station abnormalities.

LIMITING CONDITIONS FOR OPERATION

3.6.D Relief Valves

MSRV	Affected Areas/Zones
2-PCV-1-19	2-3, 2-4, 9
2-PCV-1-22	2-2
2-PCV-1-23	2-2
2-PCV-1-31	2-3, 2-4, 9
2-PCV-1-179	2-3, 2-4, 9
2-PCV-1-180	2-2

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.D. Relief Valves

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

3.6.E Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recircution loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

   b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

   c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

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.

In fire area 9 and fire zones 2-2, 2-3, and 2-4, a postulated fire could potentially disable all but three MSRVs. If one of these three MSRVs was the MSRV allowed by the technical specifications to be indefinitely out of service, then the required number of three MSRVs for safe shutdown would not be available. If one of the required MSRVs is out of service, an hourly patrolling fire watch will be established in the appropriate fire areas/zones as a compensatory measure. For a fire in any other fire areas/zones of the plant, at least four MSRVs would be available. Thus, even if one MSRV is out of service, the required number of three MSRVs would remain available for safe shutdown.

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.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.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)

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to all the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

1. The reactor shall not be started up (made critical) from the COLD CONDITION unless the following are satisfied:

   a. Diesel generators A, B, C, D, 3A, 3B, 3C and 3D OPERABLE.

   b. Requirements 3.9.A.3 through 3.9.A.6 are met.

   c. At least two of the following offsite power sources are available:

      (1) The 500-kV system is available to the units 1 and 2 shutdown boards through the unit 1 station-service transformer TUSS 1B with no credit taken for the two 500-kV Trinity lines. If the unit 2 station-service transformer is the second source, a minimum of two 500-kV lines must be available.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the periodic testing requirements of the auxiliary electrical system.

Objective

Verify the operability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical System

1. Diesel Generators

   a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a 1-hour period at 75% of rated load or greater.

      During the monthly generator test, the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and speed shall be logged.

3.9/4.9 AUXILIARY ELECTRICAL SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.9.A. Auxiliary Electrical Equipment

2. The reactor shall not be started up (made critical) from the HOT STANDBY CONDITION unless all of the following conditions are satisfied:

    a. At least one offsite power source is available as specified in 3.9.A.1.c.

    b. Three units 1 and 2 diesel generators and three unit 3 diesel generators shall be OPERABLE.

    c. An additional source of power consisting of one of the following:

        (1) A second offsite power source available as specified in 3.9.A.1.c.

        (2) A fourth OPERABLE units 1 and 2 diesel generator and a fourth OPERABLE unit 3 diesel generator.

    d. Requirements 3.9.A.3 through 3.9.A.6 are met.

SURVEILLANCE REQUIREMENTS

4.9.A. Auxiliary Electrical System

2. DC Power System - Unit Batteries (250-V), Diesel-Generator Batteries (125-V) and Shutdown Board Batteries (250-V)

    a. Every week the specific gravity, voltage and temperature of the pilot cell and overall battery voltage shall be measured and logged.

    b. Every three months the measurement shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

    c. A battery rated discharge (capacity) test shall be performed and the voltage, time, and output current measurements shall be logged at intervals not to exceed 24 months.

LIMITING CONDITIONS FOR OPERATION

3.9.A. Auxiliary Electrical Equipment

3. Buses and Boards Available

   a. The respective start bus is energized for each common station-service transformer designated as an offsite power source.

   b. The 4-kV bus tie board is energized and capable of supplying power to the units 1 and 2 shutdown boards if a cooling tower transformer is designated as an offsite power source.

   c. The units 1 and 2 and unit 3 4-kV shutdown boards are energized.

SURVEILLANCE REQUIREMENTS

4.9.A. Auxiliary Electrical System

3. Logic Systems

   a. Both divisions of the common accident signal logic system shall be tested every 6 months to demonstrate that it will function on actuation of the core spray system of each reactor to provide an automatic start signal to all 4 units 1 and 2 diesel generators.

   b. Once every 6 months, the condition under which the 480-volt load shedding logic system is required shall be simulated using pendant test switches and/or pushbutton test switches to demonstrate that the load shedding logic system would initiate load shedding signals on the diesel auxiliary boards, RMOV boards, and the 480-V shutdown boards.

LIMITING CONDITIONS FOR OPERATION

3.9.A. Auxiliary Electrical Equipment

3.9.A.3. (Cont'd)

d. The 480-V shutdown boards 1A, 2A, 2B, 3A, and 3B are energized.

e. The units 1 and 2 and unit 3 auxiliary boards are energized.

f. Loss of voltage and degraded voltage relays OPERABLE on 4-kV shutdown boards A, B, C, D, 3EA, 3EB, 3EC, and 3ED.

g. Shutdown buses 1 and 2 energized.

h. The 480-V reactor motor-operated valve (RMOV) boards 2D & 2E are energized with motor-generator (mg) sets 2DN, 2DA, 2EN, and 2EA in service.

i. The 480-V reactor motor-operated valve (RMOV) board 2C is energized.

j. The 4-kV bus tie board is available for cross-tying units 1 and 2 and unit 3 4-kV shutdown boards.

4. The three 250-V unit batteries, the four units 1 and 2 shutdown board batteries and 3EB shutdown board battery, a battery charger for each battery, and associated battery boards are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.9.A. Auxiliary Electrical System

4. Undervoltage Relays

a. (Deleted)

b. Once every 6 months, the the conditions under which the loss of voltage and degraded voltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

LIMITING CONDITIONS FOR OPERATION

3.9.A. Auxiliary Electrical Equipment

5. Logic Systems

   a. Common accident signal logic system is OPERABLE.

   b. 480-V load shedding logic system is OPERABLE.

6. Diesel Fuel

   a. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel-generator fuel tanks for units 1 and 2.

   b. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel-generator fuel tanks for units 3.

SURVEILLANCE REQUIREMENTS

4.9.A. Auxiliary Electrical System

4.9.A.4. (Cont'd)

   c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

   d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

5. 480-V RMOV Boards 2D and 2E

   a. Once per operating cycle the automatic transfer feature for 480-V RMOV boards 2D and 2E shall be functionally tested to verify auto-transfer capability.

LIMITING CONDITIONS FOR OPERATION

3.9.B. Operation with Inoperable Equipment

Whenever the reactor is in STARTUP mode or RUN mode and not in a COLD CONDITION, the availability of electric power shall be as specified in 3.9.A except as specified herein.

1. From and after the date that only one offsite power source is available, REACTOR POWER OPERATION is permissible for 7 days.

2.a From and after the date that the 4-kV bus tie board becomes INOPERABLE, REACTOR POWER OPERATION is permissible indefinitely provided one of the required offsite power sources is not supplied from the 161-kV system through the bus tie board.

2.b If the 4-kV bus tie board becomes unavailable for cross-tying units 1 and 2 and unit 3 4-kV shutdown boards, within the next hour establish a patrolling fire watch in fire zones 2-3 and 2-4 to ensure that the affected fire zones are checked hourly.

SURVEILLANCE REQUIREMENTS

4.9.B. Operation with Inoperable Equipment

1. When only one offsite power source is OPERABLE, all units 1 and 2 diesel generators must be demonstrated to be OPERABLE within 24 hours, and power availability for the associated boards shall be verified within one hour and at least once per 8 hours thereafter.

2.a When a required offsite power source is unavailable to unit 1 because the 4-kV bus tie board or a start bus is INOPERABLE, all unit 1 and 2 diesel generators shall be demonstrated OPERABLE within 24 hours, and power availability for the associated boards shall be verified within one hour and at least once per 8 hours thereafter. The remaining offsite source and associated buses shall be checked to be energized daily.

2.b No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.9.B. Operation With Inoperable Equipment

3.a When one of the units 1 and 2 diesel generator is INOPERABLE, continued REACTOR POWER OPERATION is permissible during the succeeding 7 days, provided that 2 offsite power sources are available as specified in 3.9.A.1.c and all of the unit 2 CS, RHR (LPCI and containment cooling) systems, and the remaining three units 1 and 2 diesel generators are OPERABLE. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

3.b When one unit 3 diesel generator is inoperable, continued REACTOR POWER OPERATION is permissible during the succeeding 7 days. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

4.a When one units 1 and 2 4-kV shutdown board is INOPERABLE, continued REACTOR POWER OPERATION is permissible for a period of 5 days provided that 2 offsite power sources are available as specified in 3.9.A.1.c and the remaining units 1 and 2 4-kV shutdown boards

SURVEILLANCE REQUIREMENTS

4.9.B. Operation With Inoperable Equipment

3.a When one of the units 1 and 2 diesel generators is found to be INOPERABLE, all of the CS, RHR (LPCI and containment cooling) systems and the remaining diesel generators and associated boards shall be demonstrated to be OPERABLE immediately and daily thereafter.

3.b No additional surveillance required.

4.a When one units 1 and 2 4-kV shutdown board is found to be INOPERABLE, all remaining units 1 and 2 diesel generators associated with the remaining 4-kV shutdown boards shall be demonstrated to be OPERABLE within 24

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.B. Operation With Inoperable Equipment	4.9.B. Operation With Inoperable Equipment
and associated diesel generators, and unit 2 CS, RHR (LPCI and containment cooling) systems, and all unit 2 480-V emergency power boards are OPERABLE. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.	hours and power availability for the remaining 4-kV shutdown boards shall be verified within 1 hour and at least once per 8 hours thereafter.
4.b When one unit 3 4-kV shutdown board is inoperable, continued REACTOR POWER OPERATION is permissible for a period of 5 days. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.	4.b No additional surveillance required.
5. When one of the shutdown buses is INOPERABLE, REACTOR POWER OPERATION is permissible for a period of 7 days.	5. When a shutdown bus is found to be INOPERABLE, all 1 and 2 diesel generators shall be proven OPERABLE within 24 hours.
6.a When one of the units 1 and 2 480-V diesel auxiliary boards becomes INOPERABLE, REACTOR POWER OPERATION is permissible for a period of 5 days.	6. When one units 1 and 2 diesel auxiliary board is found to be INOPERABLE, each unit 1 and 2 diesel generator shall be proven OPERABLE within 24 hours and power availability for the remaining diesel auxiliary board shall be verified within 1 hour and at least once per 8 hours thereafter.
6.b When one of the unit 3 480-V diesel auxiliary boards become INOPERABLE, REACTOR POWER OPERATION is permissible for a period of 5 days.	

LIMITING CONDITIONS FOR OPERATION

3.9.B Operation With Inoperable Equipment

7. From and after the date that one of the three 250-V unit batteries and/or its associated battery board is found to be INOPERABLE for any reason, continued REACTOR POWER OPERATION is permissible during the succeeding 7 days. Except for routine surveillance testing, NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period, and the plans to return the failed component to an OPERABLE state.

8. From and after the date that one of the 250-V shutdown board batteries and/or its associated battery board is found to be INOPERABLE for any reason, continued REACTOR POWER OPERATION is permissible during the succeeding five days in accordance with 3.9.B.7.

9. When one division of the logic system is INOPERABLE, continued REACTOR POWER OPERATION is permissible under this condition for seven days, provided the CSCS requirements listed in Specification 3.9.B.3 are satisfied. The NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period, and the plans to return the failed component to an OPERABLE state.

10. (deleted)

11. The following limiting conditions for operation exist for the undervoltage relays which start the diesel generators on the 4-kV shutdown boards.

SURVEILLANCE REQUIREMENTS

4.9.B Operation With Inoperable Equipment

6.b No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.9.B. Operation With Inoperable Equipment

3.9.B.11 (Cont'd)

a. The loss of voltage relay channel which starts the diesel generator for a complete loss of voltage on a 4-kV shutdown board may be INOPERABLE for 10 days provided the degraded voltage relay channel on that shutdown board is OPERABLE (within the surveillance schedule of 4.9.A.4.b).

b. The degraded voltage relay channel which starts the diesel generator for degraded voltage on a 4-kV shutdown board may be INOPERABLE for 10 days provided the loss of voltage relay channel on that shutdown board is OPERABLE (within the surveillance schedule of 4.9.A.4.b).

c. One of the three phase-to-phase degraded voltage relays provided to detect a degraded voltage on a 4-kV shutdown board may be INOPERABLE for 15 days provided both of the following conditions are satisfied.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.9.B. Operation With Inoperable Equipment

12. When one unit 2 480-V shutdown board is found to be INOPERABLE, the reactor will be placed in the HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

13. If one unit 2 480-V RMOV board mg set is INOPERABLE, the REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

14. If any two unit 2 480-V RMOV board mg sets become INOPERABLE, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

15. When one 480-V shutdown board (1A or 3A or 3B) is found to be INOPERABLE, REACTOR POWER OPERATION is permissible for a period of 7 days.

16. If the 480-V RMOV board 2C becomes INOPERABLE, within the next hour establish a patrolling fire watch in fire zones 2-5 and 2-6 to ensure these zones are checked hourly.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.9.B. Operation With Inoperable Equipment

17. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.16 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

3.9.C. Operation in COLD SHUTODWN

Whenever the reactor is in COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two units 1 and 2 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.

2. An additional source of power energized and capable of supplying power to the units 1 and 2 shutdown boards consisting of at least one of the following:

   a. One of the offsite power sources specified in 3.9.A.1.c.

   b. A third OPERABLE diesel generator.

3. At least one 480-V shutdown board for each unit must be OPERABLE.

4. One 480-V RMOV board mg set is required for each RMOV board (2D or 2E) required to support operation of the RHR system in accordance with 3.5.B.9.

3.9 BASES

The objective of this specification is to assure an adequate source of electrical power to operate facilities to cool the plant during shutdown, to operate the engineered safeguards following an accident, and to bring the plant to cold shutdown for a fire at any location. There are three sources of alternating current electrical energy available, namely, the 161-kV transmission system, the 500-kV transmission system, and the diesel generators.

For a fire, units 1 and 2 and unit 3 diesel generators and associated electrical distribution systems are required to be available in various combinations to ensure adequate power to safe shutdown systems. The plant Appendix R evaluation establishes the need for certain units 1 and 3 auxiliary power systems to achieve and maintain cold shutdown on unit 2. For this reason, these required systems have been added to the unit 2 technical specifications with allowed inoperable periods which are identical to the existing unit 1 and 3 technical specifications.

The unit station-service transformer B for unit 1 or the unit station-service transformer B for unit 2 provide noninterruptible sources of offsite power from the 500-kV transmission system to the units 1 and 2 shutdown boards. Auxiliary power can also be supplied from the 161-kV transmission system through the common station-service transformers or through the cooling tower transformers by way of the bus tie board. The 4-kV bus tie board may remain out of service indefinitely provided one of the required offsite power sources is not supplied from the 161-kV system through the bus tie board. For a fire, the 4-kV bus tie board is used to cross-tie the units 1 and 2 and unit 3 4-kV shutdown boards so that power from unit 3 diesel generators can be provided to unit 2 for various fire locations. As previously stated, the 4-kV bus tie board may be out of service indefinitely provided the required offsite power sources are available. However, the plant Appendix R evaluation requires that the 4-kV bus tie board cross-tie capability be available at all times. If the 4-kV bus tie board is unavailable for cross-tying, an hourly patrolling fire watch is required to be established in fire zones 2-3 and 2-4.

The minimum fuel oil requirement of 103,300 gallons is sufficient for seven days of full load operation of three units 1 and 2 diesels and is conservatively based on availability of a replenishment supply. An identical requirement is provided for the unit 3 diesels.

The degraded voltage sensing relays provide a start signal to the diesel generators in the event that a deteriorated voltage condition exists on a 4-kV shutdown board. This starting signal is independent of the starting signal generated by the complete loss of voltage relays and will continue to function and start the diesel generators on complete loss of voltage should the loss of voltage relays become inoperable. The 15-day inoperable time limit specified when one of the three phase-to-phase degraded voltage relays is inoperable is justified based on the two-out-of-three permissive logic scheme provided with these relays.

A units 1 and 2 4-kV shutdown board is allowed to be out of operation for a brief period to allow for maintenance and testing, provided all remaining units 1 and 2 4-kV shutdown boards and associated diesel generators, CS, RHR, (LPCI and containment cooling) systems supplied by the remaining units 1 and 2 4-kV shutdown boards, and all emergency 480-V power boards are operable. A unit 3 4-kV shutdown board is allowed to be out of operation for a brief period to allow for maintenance and testing.

There are eight 250-V dc battery systems, each of which consists of a battery, battery charger, and distribution equipment. Three of these systems provide power for unit control functions, operative power for unit motor loads, and alternative drive power for a 115-V ac unit-preferred mg set. One 250-V dc system provides power for common plant and transmission system control functions, drive power for a 115-V ac plant-preferred mg set, and emergency drive power for certain unit large motor loads. The four remaining systems deliver control power to the 4,160-V shutdown boards.

Each 250-V dc shutdown board control power supply can receive power from its own battery, battery charger, or from a spare charger. The chargers are powered from normal plant auxiliary power or from the standby diesel-driven generator system. Zero resistance short circuits between the control power supply and the shutdown board are cleared by fuses located in the respective control power supply. Each power supply is located in the reactor building near the shutdown board it supplies. Each battery is located in its own independently ventilated battery room.

The 250-V dc system is so arranged, and the batteries sized so that the loss of any one unit battery will not prevent the safe shutdown and cooldown of all three units in the event of the loss of offsite power and a design basis accident in any one unit. Loss of control power to any engineered safeguard control circuits is annunciated in the main control room of the unit affected. The loss of one 250-V shutdown board battery affects normal control power only for the 4,160-V shutdown board which it supplies. The station battery supplies loads that are not essential for safe shutdown and cooldown of the nuclear system. This battery was not considered in the accident load calculations.

There are two 480-V ac RMOV boards that contain mg sets in their feeder lines. These 480-V ac RMOV boards have an automatic transfer from their normal to alternate power source (480-V ac shutdown boards). The mg sets act as electrical isolators to prevent a fault from propagating between electrical divisions due to an automatic transfer. The 480-V ac RMOV boards involved provide motive power to valves associated with the LPCI mode of the RHR system. Having an mg set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required. Since sufficient equipment is available to maintain the minimum complement required for RHR (LPCI) operation, a 7-day servicing period is justified. Having two mg sets out of service can considerably reduce equipment availability; therefore, the affected unit shall be

placed in Cold Shutdown within 24 hours. 480-V RMOV Board 2C is required to be operable since it is used to supply power for specific fire locations. If 480-V RMOV Board 2C becomes inoperable, an hourly patrolling fire watch is required to be established in fire zones 2-5 and 2-6.

The offsite power source requirements are based on the capacity of the respective lines. The Trinity line is limited to supplying two operating units because of the load limitations of CSST's A and B. The Athens line is limited to supplying one operating unit because of the load limitations of the Athens line. The limiting conditions are intended to prevent the 161-kV system from supplying more than two units in the event of a single failure in the offsite power system.

The monthly tests of the diesel generators are primarily to check for failures and deterioration in the system since last use. The diesels will be loaded to at least 75 percent of rated power while engine and generator temperatures are stabilized (about one hour). The minimum 75-percent load will prevent soot formation in the cylinders and injection nozzles. Operation up to an equilibrium temperature ensures that there is no overheating problem. The tests also provide an engine and generator operating history to be compared with subsequent engine-generator test data to identify and to correct any mechanical or electrical deficiency before it can result in a system failure.

The test during refueling outages is more comprehensive, including procedures that are most effectively conducted at that time. These include automatic actuation and functional capability tests to verify that the generators can start and be ready to assume load in 10 seconds. The annual inspection will detect any signs of wear long before failure. The diesel generators are shared by units 1 and 2. Therefore, the capability for the units 1 and 2 diesel generators to accept the emergency loads will be performed during the unit 1 operating cycle using the unit 1 loads.

Battery maintenance with regard to the floating charge, equalizing charge, and electrolyte level will be based on the manufacturer's instruction and sound maintenance practices. In addition, written records will be maintained of the battery performance. The plant batteries will deteriorate with time but precipitous failure is unlikely. The type of surveillance called for in this specification is that which has been demonstrated through experience to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the ampere-hour capacity of the battery, and will be applied as recommended.

4.9 BASES

The testing of the logic systems will verify the ability of the logic systems to bring the auxiliary electrical system to running standby readiness with the presence of an accident signal from any reactor or an undervoltage signal on the 4-kV shutdown boards.

The periodic simulation of accident signals in conjunction with diesel-generator voltage available signals will confirm the ability of the 480-V load shedding logic system to sequentially shed and restart 480-V loads if an accident signal were present and diesel-generator voltage were the only source of electrical power.

The unit 3 diesel generators and associated electrical distribution systems requirements for operability and surveillance are identical to the existing unit 3 technical specifications. However, if a unit 3 diesel generator or associated electrical distribution system becomes inoperable, no additional surveillance is required. Since the Appendix R shutdown equipment powered by the remaining unit 3 power sources are not redundant to the inoperable equipment, additional testing would not improve the reliability of the power supplies for a specific fire location.

REFERENCES

1. Normal Auxiliary Power System (BFNP FSAR Subsection 8.4)

2. Standby AC Power Supply and Distribution (BFNP FSAR Subsection 8.5)

3. 250-Volt DC Power Supply and Distribution (BFNP FSAR Subsection 8.6)

4. Memorandum from Gene M. Wilhoite to H. J. Green dated December 4, 1981 (L00 811208 664) and memorandum from C. E. Winn to H. J. Green dated January 10, 1983 (G02 830112 002)

ENCLOSURE 2

DESCRIPTION AND JUSTIFICATION
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 2

Description of Change

BFN Unit 2 Technical Specifications are being revised to include additional unit 2 equipment not presently required to be operable and unit 1 and 3 equipment needed for unit 2 safe shutdown. See the attached technical specification markups for proposed changes.

Reason for Change

In accordance with 10 CFR 50.48 and 10 CFR 50, Appendix R, adequate protection of equipment is required to ensure the safe shutdown of a nuclear plant in the event of a fire at any location in the plant. In addition, Generic Letter 81-12, "Fire Protection Rule," requested that "Technical Specifications of the surveillance requirements and limiting conditions for operation for that equipment not already covered by existing Technical Specifications" be provided. The proposed changes are being made to address the limiting conditions with respect to the plant equipment which is being utilized for postfire shutdown of unit 2.

Justification for Change

A plant Appendix R evaluation was performed for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, to ensure that safe shutdown capability can be maintained during and after a fire in compliance with section III.G, III.J, and III.L of Appendix R. The Appendix R evaluation identified the minimum systems required to be operable for postfire safe shutdown and the modifications that were necessary to ensure the operability of the minimum systems. The plant Appendix R evaluation was performed assuming concurrent operation of the three Browns Ferry units and did not factor in the unavailability of equipment because of possible outage of a unit. A supplemental evaluation was also performed for only unit 2 operating. As a result, the safe shutdown capability of unit 2 depends upon equipment not directly covered in the existing technical specifications for unit 2.

The existing unit 2 technical specifications for the main steam relief valves (MSRVs), residual heat removal service water (RHRSW) pumps, and emergency equipment cooling water (EECW) pumps do not provide sufficient equipment operability for all postulated Appendix R events. Unit 2 safe shutdown capability also relies upon portions of the unit 1 and 3 auxiliary power systems, including the unit 3 diesel generators, which are not directly included in the unit 2 technical specifications. Also, the reactor water level and reactor vessel pressure instrumentation at the backup control panel as identified in the plant Appendix R evaluation do not currently have any technical specification operability requirements.

MSRVs - The existing unit 2 technical specifications for the MSRVs permit indefinite plant operation with one relief valve inoperable. The plant Appendix R evaluation assessed the availability of MSRVs to ensure that three MSRVs are available for any given fire location for safe shutdown. In fire area 9 and fire zones 2-2, 2-3, and 2-4, a postulated fire could result in only three MSRVs being available. If one of these three MSRVs was the one currently allowed by the technical specification to be indefinitely out of service, the required number of MSRVs (three) for safe shutdown during a fire would not be available. An hourly patrolling fire watch will be established in the appropriate fire areas/zones as a compensatory measure if one of the required MSRVs is out of service. For a fire in any other areas/zones of the plant, at least four MSRVs would be available. Thus, even if one were out of service, the required number of three MSRVs would still be available for safe shutdown.

The proposed technical specifications ensure a safe shutdown capability and provide a compensatory measure during plant operations with an inoperable MSRV. Establishing a patrolling fire watch within one hour is intended to observe hazardous conditions which are not normally detected by installed fire protection systems. These conditions include activities by plant personnel that could increase the hazards of a fire. They also include conditions likely to lead to a fire, such as spills of flammable liquids or the presence of ignition sources, and accumulations of transient combustible materials. The patrolling fire watch is intended to provide prompt notification of a fire and to provide fire fighting activities until the fire brigade responds. The patrolling fire watch provides assurances that the existence of unsafe or fire conditions would be minimized.

Auxiliary Power System - The existing unit 2 technical specifications for the auxiliary power system require the operability of the units 1 and 2 diesel generators and associated auxiliary power distribution systems. The associated auxiliary power distribution systems includes 4-kV and 480V shutdown boards, 480V reactor motor-operated valve (RMOV) boards, 250V unit batteries and associated chargers and boards. The auxiliary power system is required to provide a postfire power source for the plant equipment. The plant Appendix R evaluation assumed the availability of the units 1 and 2 diesel generators, unit 3 diesel generators, and associated power distribution systems. The plant Appendix R evaluation assumed the 4-kV bus tie board cross-tie capability to be available at all times to cross-tie unit 1 and 2 and unit 3 4kV shutdown boards. Additionally, since 480V RMOV board 2C supplies power to valves which are operated during a fire, this board has also been added to the technical specifications. The proposed changes are to transfer the appropriate sections of the unit 1 and unit 3 technical specifications for the auxiliary power system to the unit 2 technical specification. Presently some of the unit 3 equipment is indirectly included

in the unit 2 technical specifications through the definition of operability (e.g., a unit 3 diesel is required to be available to power an EECW pump required for unit 2 operation). The proposed changes explicitly add the unit 1 and 3 auxiliary power systems which are required to be operable for postfire safe shutdown to the unit 2 technical specifications. The unit 1 and 3 technical specifications are not affected by this change.

The shutdown requirements for inoperable unit 1 and 3 auxiliary power system components are identical to the existing unit 2 requirements. If the 4-kV bus tie board or 480-V RMOV board 2C is inoperable, a patrolling fire watch is established in fire zones 2-3 and 2-4 or zones 2-5 and 2-6, respectively. The previous discussion under MSRVs provides justification for patrolling fire watches.

The unit 3 diesel generators and associated electrical distribution systems requirements for operability and surveillance are identical to the existing unit 3 technical specifications. However, if a unit 3 diesel generator or associated electrical distribution systems becomes inoperable, no additional surveillance is required. Since the remaining unit 3 diesel generators do not supply power to the required shutdown equipment powered by the inoperable diesel generator, additional testing would not improve the reliability of the power supplies for the established Appendix R events.

The proposed technical specifications ensure adequate emergency power for postfire safe shutdown and provide a compensatory measure during plant operations with inoperable equipment.

Reactor Vessel Instrumentation - The existing unit 2 technical specifications (table 3.2.F) for instruments require the operation of reactor vessel water level and reactor pressure indication in the control room. The plant Appendix R evaluation assumed that the reactor vessel water level and pressure indicators on the backup control panel were also available for fires in the control bay that could force plant operators to abandon the main control room. The proposed technical specifications ensure a safe shutdown capability and provide a compensatory measure during plant operation with the backup control panel instruments inoperable. The compensatory measure of the patrolling fire watch provides assurances that the existence of unsafe or fire conditions would be be minimized. The previous discussion under MSRVs provides justification for patrolling fire watches. A surveillance requirement for these instruments on the backup panel is added which is identical to the requirement for the instruments in the control room.

RHRSW and EECW Systems - The existing unit 2 technical specifications for the RHRSW and EECW pumps permit indefinite plant operation with one RHRSW and one EECW pump inoperable when three units are operating. The number of RHRSW pumps required to be operable is further reduced with units 1 and 3 in a cold shutdown condition or defueled. The plant Appendix R evaluation assumed the availability of all four EECW pumps. In fire areas 9, 16, and 18, and fire zones 2-1, 2-2, 2-3, 2-4, 2-5, and 2-6, a postulated fire could result in only two EECW pumps being available that are required by the plant Appendix R evaluation. If one of these two EECW pumps was the one currently allowed by the technical specifications to be indefinitely out of service, then the required two EECW pumps for safe shutdown would not be available. For a fire in any other areas/zones of the plant, adequate RHRSW swing/EECW pumps are available to supply necessary cooling water to the diesel generators even if one of the EECW pumps is out of service.

The plant Appendix R evaluation required that either RHRSW pumps C1 or D2 be available, however both pumps are required to be operable to ensure the one required RHRSW pump is available for a specific fire location. If one of the two required RHRSW pumps is out of service, an hourly patrolling fire watch will be established in the appropriate fire areas/zones as a compensatory measure.

For postulated fires in any other areas of the plant (i.e., other than 2-1, 2-2, 2-3, 2-4, 2-5, 2-6, 9, 16, 18), one train of equipment needed to achieve and maintain hot shutdown will be free of fire damage through fire area boundary separation. In those cases where hot shutdown is assured and alternate shutdown is not required, plant operating instructions (e.g., EOIs) will be used to complete the cooldown process. The plant Appendix R evaluation for Unit 2 operation further identified equipment which can be used to reach cold shutdown without repair. With hot shutdown assured, adequate time is available for the operators to perform necessary actions using symptom oriented procedures (e.g., EOIs) to ensure that there are adequate RHRSW and EECW pumps available to achieve cold shutdown. This will provide the flexibility to align equipment which may be operable but not necessarily a preselected shutdown path.

The proposed technical specifications ensure a safe shutdown capability by providing a compensatory measure during plant operation with inoperable RHRSW and EECW pumps. The compensatory measure of the patrolling fire watch provides assurances that the existence of unsafe or fire conditions would be minimized. The previous discussion under MSRVs provides justification for patrolling fire watches.

ENCLOSURE 3

BROWNS FERRY NUCLEAR PLANT
SIGNIFICANT HAZARDS CONSIDERATION
UNIT 2

Description of Amendment Request

The proposed amendment would change the technical specifications of Browns Ferry Nuclear Plant Unit 2 by revising the limiting conditions for operation, the surveillance requirements, and periodicity for equipment required for Appendix R safe shutdown.

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, (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 amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendment does not alter the function or the testing of any equipment or systems previously analyzed in the BFN Final Safety Analysis Report, but provides additional equipment operability requirements to support the safe shutdown of the plant for a fire event.

2. The proposed amendment does not create the possibility of a new or different kind of accident from an accident previously evaluated. This proposed change is still within the bounds of the design of the systems. Equipment previously covered by units 1 and 3 technical specifications are incorporated into the unit 2 technical specifications to ensure availability to support unit 2 safe shutdown during a fire for periods when units 1 and 3 may be shutdown.

3. The proposed amendment does not involve a significant reduction in the margin of safety. The proposed change ensures a safe shutdown capability for a fire at any location in the plant. It does not alter the safety function of the involved equipment.

Determination of Basis for Proposed No Significant Hazards

Since the application for amendment involves a proposed change that is encompassed by the criteria for which no significant hazards consideration exists, TVA has made a proposed determination that the application involves no significant hazards consideration.

3.5.L. APRM Setpoints

Operation is constrained to a maximum LHGR of 18.5 kW/ft for 7x7 fuel and 13.4 kW/ft. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2, NEDO - 24088-1 and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE - 24011-P-A and Addenda.

3.5 BASES (Cont'd)

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier. Note that this specification applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures yet be fully capable of performing their pressure relief function.
Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

With two ADS valves known to be incapable of automatic operation, four valves remain operable to perform their ADS function. The ECCS loss-of-coolant accident analyses for small line breaks assumed that four of the six ADS valves were operable. Reactor operation with three ADS valves inoperable is allowed to continue for seven days provided that the HPCI system is demonstrated to be operable. Operation with more than three of the six ADS valves inoperable is not acceptable.

3.5.H. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, HPCIS, and RCICS are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for Technical Specification purposes.

The core spray and RHR system discharge piping highpoint vent is visually checked for water flow once a month prior to testing to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a pressure suppression chamber head tank is located approximately 20 feet above the discharge line highpoint to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge highpoint serves as a backup charging system when the pressure suppression chamber head tank is not in service. System discharge pressure indicators are used to determine the water level above the discharge line highpoint. The indicators will reflect approximately 30 psig for a water level at the highpoint and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the HPCI and RCIC pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the HPCIS and RCICS piping. This assures that the HPCI and RCIC discharge piping remains filled. Further assurance is provided by observing water flow from these systems' highpoints monthly.

3.5.I. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.
The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1 and -2. The analyses supporting these limiting values are presented in Reference 1.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent rated thermal power, the R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

LIMITING CONDITIONS FOR OPERATION

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.

3.6.D. Relief Valves

1. When more than one relief valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. During REACTOR POWER OPERATION, if one of the following relief valves is inoperable, establish within the next hour a patrolling fire watch to ensure that the affected fire areas/zones listed below are checked hourly.

SURVEILLANCE REQUIREMENTS

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.

4.6.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. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.