ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION REVISIONS
(TVA BFNP TS 175)
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

UNIT 1

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekl When Required to be Operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Once/month
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/Week

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be operable or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. **(deleted)**

6. The functional test of the flow bias network is performed in accordance with Table 4.2.C.

3.6.A Thermal and Pressurization Limitations

3. During heatup by non-nuclear means, except when the vessel is vented or as indicated in 3.6.A.4, cooldown following nuclear shutdown on low-level physics tests, the reactor vessel temperatures shall be at or above the temperatures of curve #2 of figure 3.6.1.

4. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of figure 3.6-1. The applicability of this curve to these tests is extended to non-nuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

4.6.A Thermal and Pressurization Limitations

3. **Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed inthe reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-70. Samples shall be withdrawn at one-fourth and three-fourths service life.**

4. Neutron flux wires which were installed adjacent to the reactor vessel wall at the core midplane level were removed during the first refueling outage and tested. The results were used to more accurately determine the neutron fluence in the vessel beltline shell at a depth of one-fourth of the wall thickness. These determined values of neutron fluence and the methods in Regulatory Guide 1.99 were used to predict the changes in reference temperature, RTNDT, for each effective full power year. Figure 3.6-1 shall be updated based on these predictions until the results from testing the specimens described in 4.6.A.3 are available.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

3.6.C Coolant Leakage

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

D. Relief Valves

1. When more than one relief valve or one or more safety valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

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 shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. Relief Valves

1. At least one safety valve and approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves (2 safety and 11 relief) 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.

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

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

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

DELETED

FIGURE 3.6-2
CHANGE IN CHARPY V TRANSITION TEMPERATURE
VERSUS
NEUTRON EXPOSURE

3.6/4.6 BASES

3.6.A/4.6.A Thermal and Pressurization Limitations

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100° F per hour were considered in the temperature range of 100 to 546° F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (65 Edition including Summer 1966 addenda).

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating Limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. For the purpose of setting these operating limits the reference temperature, RTNDT, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (65 Edition to Summer 1966 addenda.)

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: 1) A relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux), and b) a measure of the neutron fluence at the point of interest in the RPV wall.

A relationship between neutron fluence and change in reference temperature, RT_{NDT}, is provided in Regulatory Guide 1.99. In turn, this change in reference temperature can be related to a change in the temperature ordinate shown in Figure G-2110-1 in Appendix G of Section III of ASME Boiler and Pressure Vessel Code.

The change in reference temperature at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Regulatory Guide 1.99. During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which had been installed adjacent to the vessel wall at the core midplane level were removed and tested to determine the neutron fluence. Three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone have also been placed adjacent to the vessel wall at the core midplane level. These will be removed and tested as required by 10CFR50, Appendix H. Until such testing is performed, the changes in reference temperature, RT_{NDT}, will be determined based on the results of the testing of the dosimeter wires and the methods in Regulatory Guide 1.99. The operating pressure-temperature

3.6/4.6 BASES

3.6.A/4.6.A

limits shown in Figure 3.6-1 will be adjusted periodically to conservatively account for these determined changes in the reference temperature.

As described in paragraph 4.2.5 of the safety analysis report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temerature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperaure than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The requirements for full tension boltup of the reactor vessel closure are based on the NDT temperature plus 60°F. This is derived from the requirements of the ASME code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head, and shell material is a maximum of 40°F and a maximum of 10°F for the stud material. Therefore, the minimum temperature for full tension boltup is 40°F plus 60°F for a total of 100°F. The partial boltup is restricted to the full loading of eight studs at 70°F, which is stud NDT temperature (10°F) plus 60°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt $\geq$ 1 Mev; therefore, radiation effects will be minor and will not influence this temperature.

3.6.B/4.6.B Coolant Chemistry

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

3.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be operable at all times when secondary containment integrity is required.

4.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. At least once per year, the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm (± 10%).

b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975 and are capable of an output of at least 40 kW.

c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

.7 CONTAINMENT SYSTEMS

E. Control Room Emergency Ventilation

1. Except as specified in specification 3.7.E.3 below, both control room emergency pressurization systems shall be operable at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.).

4.7 CONTAINMENT SYSTEMS

E. Control Room Emergency Ventilation

1. At least once per operating cycle, not to exceed 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

UNIT 2

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekl When Required to be Operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Once/month
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/Week

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be operable or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. Deleted

6. The functional test of the flow bias network is performed in accordance with Table 4.2.C.

3.6.A Thermal and Pressurization Limitations

3. During heatup by non-nuclear means, except when the vessel is vented or as indicated in 3.6.A.4, cooldown following nuclear shutdown on low-level physics tests, the reactor vessel temperatures shall be at or above the temperatures of curve #2 of figure 3.6.1.

4. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of figure 3.6-1. The applicability of this curve to these tests is extended to non-nuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

4.6.A Thermal and Pressurization Limitations

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed inthe reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-70. Samples shall be withdrawn at one-fourth and three-fourths service life.

4. Neutron flux wires which were installed adjacent to the reactor vessel wall at the core midplane level were removed during the first refueling outage and tested. The results were used to more accurately determine the neutron fluence in the vessel beltline shell at a depth of one-fourth of the wall thickness. These determined values of neutron fluence and the methods in Regulatory Guide 1.99 were used to predict the changes in reference temperature, RTNDT, for each effective full power year. Figure 3.6-1 shall be updated based on these predictions until the results from testing the specimens described in 4.6.A.3 are available.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

3.6.C Coolant Leakage

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

D. Relief Valves

1. When more than one relief valve or one or more safety valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

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 shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. Relief Valves

1. At least one safety valve and approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves (2 safety and 11 relief) 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.

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

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

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

DELETED

FIGURE 3.6-2
CHANGE IN CHARPY V TRANSITION TEMPERATURE
VERSUS
NEUTRON EXPOSURE

3.6/4.6 BASES

3.6.A/4.6.A Thermal and Pressurization Limitations

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100° F per hour were considered in the temperature range of 100 to 546° F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (65 Edition including Summer 1966 addenda).

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. For the purpose of setting these operating limits the reference temperature, RTNDT, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (65 Edition to Summer 1966 addenda.)

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: 1) A relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux), and b) a measure of the neutron fluence at the point of interest in the RPV wall.

A relationship between neutron fluence and change in reference temperature, RT_{NDT}, is provided in Regulatory Guide 1.99. In turn, this change in reference temperature can be related to a change in the temperature ordinate shown in Figure G-2110-1 in Appendix G of Section III of ASME Boiler and Pressure Vessel Code.

The change in reference temperature at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Regulatory Guide 1.99. During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which had been installed adjacent to the vessel wall at the core midplane level were removed and tested to determine the neutron fluence. Three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone have also been placed adjacent to the vessel wall at the core midplane level. These will be removed and tested as required by 10CFR50, Appendix H. Until such testing is performed, the changes in reference temperature, RT_{NDT}, will be determined based on the results of the testing of the dosimeter wires and the methods in Regulatory Guide 1.99. The operating pressure-temperature

3.6/4.6 BASES

3.6.A/4.6.A

limits shown in Figure 3.6-1 will be adjusted periodically to conservatively account for these determined changes in the reference temperature.

As described in paragraph 4.2.5 of the safety analysis report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temerature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperaure than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The requirements for full tension boltup of the reactor vessel closure are based on the NDT temperature plus 60°F. This is derived from the requirements of the ASME code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head, and shell material is a maximum of 40°F and a maximum of 10°F for the stud material. Therefore, the minimum temperature for full tension boltup is 40°F plus 60°F for a total of 100°F. The partial boltup is restricted to the full loading of eight studs at 70°F, which is stud NDT temperature (10°F) plus 60°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt $\geq$ 1 Mev; therefore, radiation effects will be minor and will not influence this temperature.

3.6.B/4.6.B Coolant Chemistry

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

3.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be operable at all times when secondary containment integrity is required.

4.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. At least once per year, the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm (± 10%).

b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975 and are capable of an output of at least 40 kW.

c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

.7 CONTAINMENT SYSTEMS

E. Control Room Emergency Ventilation

1. Except as specified in specification 3.7.E.3 below, both control room emergency pressurization systems

shall be operable at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.).

4.7 CONTAINMENT SYSTEMS

E. Control Room Emergency Ventilation

1. At least once per operating cycle, not to exceed 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

UNIT 3

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Once/Month
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be operable or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. (deleted)

6. The functional test of the flow bias network is performed in accordance with Table 4.2.C.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

4. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve Number 1 of figure 3.6-1. The applicability of this curve to these tests is extended to non-nuclear heatup and ambient loss cool-down associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and bottom head drain are within 145°F.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

4. Neutron flux wires which were installed adjacent to the reactor vessel wall at the core midplane level were removed during the first refueling outage and tested. The results were used to more accurately determine the neutron fluence in the vessel beltline shell at a depth of one-fourth of the wall thickness. These determined values of neutron fluence and the methods in Regulatory Guide 1.99 were used to predict the changes in reference temperature, RTNDT, for each effective full power year. Figure 3.6-1 shall be updated based on these predictions until the results from testing the specimens described in 4.6.A.3 are available.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

3.6 PRIMARY SYSTEM BOUNDARY

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.

4.6 PRIMARY SYSTEM BOUNDARY

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.

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

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

DELETED

FIGURE 3.6-2
CHANGE IN CHARPY V TRANSITION TEMPERATURE
VERSUS
NEUTRON EXPOSURE

3.6/4.6 BASES

3.6.A/4.6.A Thermal and Pressurization Limitations

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (65 Edition including Summer 1966 addenda).

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessl shell remote from discontinuities. For the purpose of setting these operating limits the reference temperature, RTNDT, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (65 Edition to Summer 1966 addenda.).

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: 1) A relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux), and b) a measure of the neutron fluence at the point of interest in the RPV wall.

A relationship between neutron fluence and change in reference temperature, RTNDT, is provided in Regulatory Guide 1.99. In turn, this change in reference temperature can be related to a change in the temperature ordinate shown in Figure G-2110-1 in Appendix G of Section III of the ASME Boiler and Pressure Vessel Code.

The change in reference temperature at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Regulatory Guide 1.99. During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which had been installed adjacent to the vessel wall at the core midplane level were removed and tested to determine the neutron fluence. Three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone have also been placed adjacent to the vessel wall at the core midplane level. These will be removed and tested as required by 10 CFR 50, Appendix H. Until such testing is performed, the changes in

reference temperature, RTNDT, will be determined based on the results of the testing of the dosimeter wires and the methods in Regulatory Guide 1.99. The operating pressure-temperature limits shown in Figure 3.6-1 will be adjusted periodically to conservatively account for these determined changes in the reference temperature.

As described in paragraph 4.2.5 of the safety analysis report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The requirements for full tension boltup of the reactor vessel closure are based on the NDT temperature plus 60°F. This is derived from the requirements of the ASME code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head, and shell material is a maximum of 40°F and a maximum of 10°F for the stud material. Therefore, the minimum temperature for full tension boltup is 40°F plus 60°F for a total of 100°F. The partial boltup is restricted to the full loading of eight studs at 70°F, which is stud NDT temperature (10°F) plus 60°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt $\geq$ 1 Mev; therefore, radiation effects will be minor and will not influence this temperature.

3.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be operable at all times when secondary containment integrity is required.

4.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment System

1. At least once per year, the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm (± 10%).

b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975 and are capable of an output of at least 40 kW.

c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

3.7 CONTAINMENT SYSTEMS

E. Control Room Emergency Ventilation

1. Except as specified in specification 3.7.E.3 below, both control room emergency pressurization systems shall be operable at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.).

4.7 CONTAINMENT SYSTEMS

E. Control Room Emergency Ventilation

1. At least once per operating cycle, not to exceed 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

ENCLOSURE 2

DESCRIPTION AND JUSTIFICATION
(TVA BFNP TS 715)
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

1. Pages 35 and 37 - Units 1 and 2
   Pages 36 and 38 - Unit 3

   Description

   The proposed changes remove the requirement to perturb the water level in the reactor vessel and monitor the water level indicator changes after performing the monthly functional test. The functional test will continue to be performed monthly as required by technical specification Table 4.1.A.

   Reason/Justficiation

   The water level instrumentation is valved out during performance of the monthly functional test. After completion of that test the level instrument is valved back into service. That instrument then indicates the reactor vessel water level. This indication can be compared with the numerous other water level instruments for verification that the instrument has indeed been returned to service.

   Perturbing the reactor water level is an operational inconvenience to the plant staff. We are not aware of any regulatory requirement or recommendation to perturb the water level. The BWR Standard Technical Specifications do not require it. The FSAR states that for any sensor that is valved-out or otherwise removed from service during testing, positive indication is obtained that the sensor has been returned to service and will see changes in the process variable. Indication of the reactor vessel water level after the instrument is valved back into service and general agreement with the other instrumentation as discussed above demonstrates compliance with the FSAR statement. Additionally, removal of this requirement will not adversely affect the operation, safety margins, accident analysis, or overall safety of the plant.

2. Page 175 - Units 1 and 2
Page 186 - Unit 3

Description

This change only updates technical specification 4.6.A.4 to reflect the present status of the neutron flux wires.

Reason

These wires were installed for purposes of experimentally verifying the calculated values of neutron fluence. This proposed change reflects that the wires were removed and the results used to determine the neutron fluence. The new proposed specification outlines planned actions regarding future revisions to the technical specifications concerning reference temperature RT_{NDT}.

No additional justification is needed.

3. Page 181 - Units 1 and 2
Page 192 - Unit 3

Description

This proposed change revises the surveillance requirement concerning monitoring of relief valve bellows to clarify that the bellows will be monitored when valves incorporating the bellows design are installed. For unit 3 this surveillance requirement had been removed but is now proposed to be added back to have all three units consistent in technical specification requirements.

Reason/Justification

All bellows type relief and safety valves have been changed out to nonbellows two-stage Target Rock valves. This proposed change only clarifies requirement for testing the integrity of bellows if spare valves with bellows are installed. The test requirement is not applicable unless a bellows type valve were to be reinstalled.

4. Page 195 - Units 1 and 2
Page 208 - Unit 3

Description

Figure 3.6-2 "Change in Charpy V Transition Temperature versus Neutron Exposure" is to be deleted.

Reason/Justification

Figure 3.6-2 as shown in the technical specifications is not consistent with Regulatory Guide 1.99. We do not consider it appropriate to have this figure in the technical specifications. As necessary in the future, the regulatory guide can be used in lieu of this Figure 3.6-2.

5. Pages 215 and 216 - Units 1 and 2
Pages 220 and 221 - Unit 3

Description

These proposed changes revise the BASES to more accurately reflect current industry practices regarding determination of changes in reference temperature RT_{NDT}. The change updates the specification BASES to reflect what has been done with neutron dosimeter wires that were installed adjacent to the reactor vessel wall. It also describes what will be done with mechanical test specimens. It describes our future plans for determining changes in reference temperature RT_{NDT}.

6. Pages 236 and 244 - Units 1 and 2
Pages 247 and 256 - Unit 3

Description

The proposed changes remove the specific references to the diesel generators required for operation of the Standby Gas Treatment System (SGTS) and the Control Room Emergency Ventilation from these sections.

Reason/Justification

These proposed changes are for clarity only. Rather than addressing diesels in the section Limiting Condition for Operation, it has been placed in the Definitions section per NRC request. (Reference letter from D. G. Eisenhut to All Power Reactor Licensees dated April 10, 1980.) The requirement for operability of the backup power supply to the SGTS and Control Room Emergency Ventilation is now addressed by technical specification definitions 1.C.2 and 1.E. Therefore, these requirements do not need to be addressed separately in the LCO. This proposed change does not affect the safe operation of the plant.

6. Pages 236 and 244 - Units 1 and 2
Pages 247 and 256 - Unit 3

Description

The proposed changes remove the specific references to the diesel generators required for operation of the Standby Gas Treatment System (SGTS) and the Control Room Emergency Ventilation from these sections.

Reason/Justification

These proposed changes are for clarity only. Rather than addressing diesels in the section Limiting Condition for Operation, it has been placed in the Definitions section per NRC request. (Reference letter from D. G. Eisenhut to All Power Reactor Licensees dated April 10, 1980.) The requirement for operability of the backup power supply to the SGTS and Control Room Emergency Ventilation is now addressed by technical specification definitions 1.C.2 and 1.E. Therefore, these requirements do not need to be addressed separately in the LCO. This proposed change does not affect the safe operation of the plant.