ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, and 3

REVISION TO TECHNICAL SPECIFICATION (TS) BASES (TS-348)
RESIDUAL HEAT REMOVAL SYSTEM INTERLOCK INSTALLATION

I. REASON FOR NOT REVISING BASES TO DISCUSS INSTALLATION OF INTERLOCKS

In Reference 1, to support a proposed amendment to the Units 1, 2, and 3 TS, TVA committed to install electrical interlocks in the Unit 1 and Unit 3 Residual Heat Removal System (RHRS) prior to each unit's restart. The interlock would be installed between the RHRS shutdown cooling pump suction valves and the corresponding suppression pool return line valves. In Reference 2, NRC requested that TVA revise the Bases for Units 1 and 3 to document the need to install these interlocks. TVA considers that adding the requested information to the Units 1 and 3 Bases is not necessary to ensure that these interlocks are installed.

First, TVA is tracking this commitment through its commitment tracking program. This ensures that the commitment will be properly dispositioned.

Second, as a test case, TVA evaluated deletion of this commitment as part of its commitment management program.[1] TVA's evaluation determined that deleting the commitment would negatively impact the ability of a system, structure, or component to perform its safety function. Therefore, the commitment will remain in effect.

Third, for restart of Units 1 and 3, TVA will employ the system plant operability checklist (SPOC) process used successfully during Unit 2 restart. SPOC is a systematic method to ensure that items affecting system operability, such as commitments, are resolved (completed) prior to declaring the system operable. Since this commitment is tied to the RHRS, TVA cannot declare the RHRS operable until this commitment is completed.

Accordingly, TVA considers that existing BFN programs provide adequate assurance that this commitment will be

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[1]TVA is participating in an industry pilot program on commitment management. This program allows TVA to revise or delete commitments which have no safety significance.

properly dispositioned, and the interlocks installed prior to each respective unit's fuel load. Furthermore, adding a commitment to the bases would necessitate two future bases changes to delete these commitments after completion.

II. REFERENCES

1. TVA letter to NRC dated January 21, 1994, Additional Information Regarding Technical Specification (TS) TS-328

2. NRC letter to TVA dated April 19, 1994, Issuance of TS Amendments for BFN Units 1, 2, and 3 Revising RHRS Operability Requirements (TS-328) (TAC Nos. M85255, M85256, and M85257)

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, and 3

REVISION TO TECHNICAL SPECIFICATION (TS) BASES (TS-348)
DESCRIPTION OF AND REASON FOR CHANGES

I. DESCRIPTION OF THE CHANGES

A. **Bases Section 2.1** - TVA is revising this section of the Units 1, 2, and 3 Bases to make editorial changes that delete unnecessary information. These changes are provided below.

1. Units 1, 2, and 3 page 1.1/2.1-11, delete the last sentence in the first paragraph (the sentence discussing that 3293 MWt is the licensed maximum power level).

2. Unit 1 page 1.1/2.1-16, Bases Section 2.1.L, delete References 3 and 4.

B. **Bases Section 3.2** - TVA is revising this section of the Units 1, 2, and 3 Bases to make a minor editorial change (Item 1) and to clarify the function of the main steam line space high temperature detection instrumentation (Item 2). These changes are listed below.

1. Unit 1 page 3.2/4.2-66, the fifth paragraph, Unit 2 page 3.2/4.2-66, the fourth full paragraph, and Unit 3 page 3.2/4.2-65, the fifth paragraph, provide a discussion of the high steam flow instrumentation. The second sentence of each paragraph currently reads:

   > The primary function of the instrumentation is to detect a break in the main steam line.

   This sentence is revised to read as follows:

   > The primary function of the high steam flow instrumentation is to detect a break in the main steam line.

2. Unit 1 page 3.2/4.2-66, starting with the last paragraph and carrying over to page 3.2/4.2-67; Unit 2 page 3.2/4.2-66, starting with the fifth full paragraph and carrying over to page 3.2/4.2-67; and Unit 3 page 3.2/4.2-65, starting with the

last paragraph and carrying over to page 3.2/4.2-66, currently read (may be more than one paragraph):

> Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of . . .

This text is revised to read as follows:

> Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks or small breaks in the main steam lines. The trip setting of 200°F for the main steam line tunnel detector is low enough to provide early indication of a steam line break. Exceeding the trip setting causes closure of isolation valves. For large breaks, the high steam tunnel temperature detection instrumentation is a backup to the high steam flow instrumentation.
>
> In the event of a loss of . . .

C. **Bases Section 3.5.F** - TVA is revising this section of the Units 1, 2, and 3 Bases to clarify the function of the Reactor Core Isolation Cooling System (RCICS). These changes are listed below.

1. Unit 1 page 3.5/4.5-31, Unit 2 page 3.5/4.5-29, and Unit 3 page 3.5/4.5-32, the first sentence currently reads:

> The RCICS functions to provide core cooling and makeup water to the reactor vessel during shutdown and isolation from the main heat sink and for certain pipe break accidents.

The revised sentence reads as follows:

> The RCICS functions to provide makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources.

2. Unit 1 page 3.5/4.5-31, Unit 2 page 3.5/4.5-29, and Unit 3 page 3.5/4.5-32, the third sentence currently reads:

> Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is required to provide core cooling.

The revised sentence reads as follows:

> Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is needed to maintain sufficient coolant to the reactor vessel.

D. **Bases Section 3.10.B** - TVA is revising this section of the Units 1, 2, and 3 Bases to make a minor editorial change and to incorporate a discussion of "fueled region." These changes are discussed below.

1. Units 1 and 2 page 3.10/4.10-13, and Unit 3 page 3.10/4.10-12, the first paragraph reads:

> The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two OPERABLE SRMs (FLCs) one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations.

This paragraph is revised to read as follows:

> The SRMs are provided to monitor the core during periods of unit shutdown and to guide the operator during refueling operations and unit startup. Requiring two OPERABLE SRMs (FLCs) during CORE

ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

Additionally, the remainder of the paragraph beginning with the third sentence (i.e., "Each SRM . . .") becomes a new paragraph.

II. REASON FOR THE CHANGES

A. **Bases Section 2.1** - TVA is revising Units 1, 2, and 3 Bases Section 2.1 (Item 1) to delete unnecessary information discussing the licensed maximum power level, since this information is provided in the operating license (License Condition 2.C.(1)). TVA is revising Unit 1 Bases Section 2.1.L (Item 2) to delete two unnecessary references.

B. **Bases Section 3.2** - TVA is revising this section of the Units 1, 2, and 3 Bases to make a minor editorial change (Item 1), and to clarify the function of the main steam line space high temperature detection instrumentation (Item 2). The revision also deletes inaccurate information and makes the bases consistent with the BFN Updated Final Safety Analysis Report (UFSAR) and BFN design documents.

UFSAR Sections 7.3.4.7 and 7.3.4.8 provide a discussion of the basis for the 200°F main steam line space high temperature trip setting and the function of the main steam line space high temperature detection instrumentation. Specifically, UFSAR Section 7.3.4.7 states that the main steam line space high temperature trip setting is set high enough to avoid spurious isolation but low enough to provide early indication of a steam line break. UFSAR Section 7.3.4.8 states that the main steam line space high temperature detection instrumentation is designed to detect leaks of from one percent to ten percent of rated steam flow.

Furthermore, neither the BFN UFSAR nor BFN design documents support the existing bases statement that the main steam line space high temperature instrumentation can detect leaks as small as 15 gallons per minute (gpm).

For large breaks, the main steam line space high temperature detection instrumentation senses the break and initiates main steam isolation valve (MSIV) closure within ten seconds, while the high flow sensors will initiate MSIV closure within 0.5 seconds. Additionally, for main steam line break accidents, BFN's accident analysis (BFN UFSAR Chapter 14) assumes the MSIVs will close on high flow. The accident analysis does not assume MSIV closure on main steam line space high temperature. Accordingly, for large breaks, the high temperature instrumentation is a backup to the high flow instrumentation.

C. **Bases Section 3.5.F** - TVA is revising this section of the Units 1, 2, and 3 Bases to clarify the function of the Reactor Core Isolation Cooling System (RCICS). This revision also makes the bases consistent with the BFN UFSAR.

UFSAR Sections 1.6.1.3.6, 4.7.1, and 7.3.4.1 discuss the function of the RCICS, and the systems needed for post-accident mitigation. Specifically, UFSAR Sections 1.6.1.3.6 and 4.7.1 state that the function of the RCICS is to provide makeup water to the reactor vessel to supplement or replace the normal makeup sources. UFSAR Section 7.3.4.1 states that, while the RCICS is expected to operate for post-accident mitigation, the RCICS does not provide any accident mitigation or safety-related function.

D. **Bases Section 3.10.B** - TVA is revising this section of the Units 1, 2, and 3 Bases in response to an NRC request. Specifically, in Reference 1, which approved amendments to the BFN Units 1, 2, and 3 TS, NRC requested that TVA revise the bases to incorporate a discussion of "fueled region." NRC also requested that the discussion of "fueled region" be consistent with the information TVA provided to NRC in Reference 2.

The revised bases are consistent with the definition of "fueled region" found in Reference 2. The revised bases are also consistent with the discussion of "fueled region" found in NRC's safety evaluation of the TS change (Reference 1).

IV. REFERENCES

1. NRC letter to TVA dated April 9, 1993, Issuance of TS Amendments for BFN Units 1, 2, and 3 Regarding Refueling Interlocks and Core Monitoring (TS-324) (TAC Nos. M84699, M84700, and M84701)

2. TVA letter to NRC dated March 31, 1993, Clarification of "Fueled Region" in TS-324

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

REVISION TO TECHNICAL SPECIFICATION (TS) BASES (TS-348)
MARKED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
1.1/2.1-11	1.1/2.1-11	1.1/2.1-11
1.1/2.1-16	3.2/4.2-66	3.2/4.2-65
3.2/4.2-66	3.2/4.2-67	3.2/4.2-66
3.2/4.2-67	3.5/4.5-29	3.5/4.5-32
3.5/4.5-31	3.10/4.10-13	3.10/4.10-12
3.10/4.10-13		

II. MARKED PAGES

See attached.

2.1 <u>BASES</u>: <u>LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY</u>

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1. ~~In addition, 3293 MWt is the licensed maximum power level for each Browns Ferry Nuclear Plant unit, and this represents the maximum steady-state power which shall not be knowingly exceeded.~~

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

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F. (Deleted)

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I.J.& K. Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. Supplemental Reload Licensing Report of Browns Ferry Nuclear Plant, Unit 1 (applicable cycle-specific document).

2. GE Standard Application for Reactor Fuel, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved version).

3. ~~"Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactor," NEDO-24154-P, October 1978.~~

4. ~~Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information On ODYN Computer Model," September 5, 1980.~~

DELETI

The low reactor water level instrumentation that is set to trip when reactor water level is 378 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

high steam flow

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks ~~in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves.~~

or small breaks in the main steam lines.

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3.2 BASES (Cont'd) SEP 2 7 1994

The trip ~~setting~~ of 200°F for the main steam line tunnel detector is low enough to ~~detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks.~~ For large breaks, the ~~high steam flow~~ high steam flow instrumentation is a backup to the ~~temperature~~ tunnel temperature detection instrumentation. NEW ¶ In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

INSERT "A" FOR BFN UNIT 1 BASES PAGE 3.2/4.2-67

provide early indication of a steam line break. Exceeding the trip setting causes closure of isolation valves.

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3.5.F Reactor Core Isolation Cooling System (RCICS)

to supplement or replace the normal makeup sources.

needed to maintain sufficient coolant to the reactor vessel.

The RCICS functions to provide ~~core cooling and~~ makeup water to the reactor vessel during shutdown and isolation from the main heat sink ~~and for certain pipe break accidents.~~ The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is ~~required to provide core cooling.~~ RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. 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.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G 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.

3.10 BASES (Cont'd)

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3.10.A (Cont'd)

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

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B. Core Monitoring

The SRMs are provided to monitor the core during periods of ~~station~~ unit shutdown and to guide the operator during refueling operations and ~~station~~ unit startup. Requiring two OPERABLE SRMs (FLCs) ~~one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations.~~

NEW ¶

Each SRM (FLC) is not required to read ≥ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

INSERT "B" FOR BFN UNIT 1 BASES PAGE 3.10/4.10-13

during CORE ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1. ~~In addition, 3293 MWt is the licensed maximum power level for each Browns Ferry Nuclear Plant unit, and this represents the maximum steady state power which shall not be knowingly exceeded.~~

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

high steam flow

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

COMBINE INTO ONE ¶

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks or small breaks in the main steam lines. ~~in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves.~~

The trip setting of 200°F for the main steam line tunnel detector is low enough to ~~detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks.~~ For large breaks, the high steam

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INSERT "C" FOR BFN UNIT 2 BASES PAGE 3.2/4.2-66

provide early indication of a steam line break. Exceeding the trip setting causes closure of isolation valves.

high steam flow

SEP 2 7 1994 NEW

tunnel temperature detection

~~flow~~ instrumentation is a backup to the ~~temperature~~ instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system. Each trip system consists of two elements. Each channel contains one temperature switch located in the pump room and three temperature switches located in the torus area. The RCIC high flow and high area temperature sensing instrument channels are arranged in the same manner as the HPCI system.

The HPCI high steam flow trip setting of 90 psid and the RCIC high steam flow trip setting of 450" H_2O have been selected such that the trip setting is high enough to prevent spurious tripping during pump startup but low enough to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

The HPCI and RCIC steam line space temperature switch trip settings are high enough to prevent spurious isolation due to normal temperature excursions in the vicinity of the steam supply piping. Additionally, these trip settings ensure that the primary containment isolation steam supply valves isolate a break within an acceptable time period to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

High temperature at the Reactor Water Cleanup (RWCU) System in the main steam valve vault, RWCU pump room 2A, RWCU pump room 2B, RWCU heat exchanger room or in the space near the pipe trench containing RWCU piping could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

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3.5.F Reactor Core Isolation Cooling System (RCICS)

to supplement or replace the normal makeup sources.

needed to maintain sufficient coolant to the reactor vessel.

The RCICS functions to provide ~~core cooling and~~ makeup water to the reactor vessel during shutdown and isolation from the main heat sink ~~and for certain pipe break accidents.~~ The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is ~~required to provide core cooling.~~ RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. 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.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G 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.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were operable. By requiring six

3.10 **BASES** (Cont'd) APR 0 9 1993

3.10.A (Cont'd)

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

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B. Core Monitoring

The SRMs are provided to monitor the core during periods of ~~station~~ unit shutdown and to guide the operator during refueling operations and ~~station~~ unit startup. Requiring two OPERABLE SRMs (FLCs) ~~one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations.~~

NEW ¶

Each SRM (FLC) is not required to read ≥ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

INSERT "D" FOR BFN UNIT 2 BASES PAGE 3.10/4.10-13

during CORE ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

2.1 <u>BASES:</u> <u>LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY</u>

MAY 2 0 1993

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1. ~~In addition, 3293 MWt is the licensed maximum power level for each Browns Ferry Nuclear Plant unit, and this represents the maximum steady state power which shall not be knowingly exceeded.~~

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

The low reactor water level instrumentation that is set to trip when reactor water level is 378 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

high steam flow

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

or small breaks in the main steam lines.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks ~~in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves.~~

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high steam flow

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3.2 BASES (Cont'd)

The trip ~~setting~~ of 200°F for the main steam line tunnel detector is low enough to ~~detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks.~~ For large breaks, the high steam ~~flow~~ instrumentation is a backup to the ~~temperature~~ tunnel temperature detection instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

NE. ¶

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" water for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

INSERT "E" FOR BFN UNIT 3 BASES PAGE 3.2/4.2-66

provide early indication of a steam line break. Exceeding the trip setting causes closure of isolation valves.

3.5.F Reactor Core Isolation Cooling System (RCICS)

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to supplement or replace the normal makeup sources.

The RCICS functions to provide ~~core cooling and~~ makeup water to the reactor vessel during shutdown and isolation from the main heat sink ~~and for certain pipe break accidents.~~ The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is ~~required to provide core cooling.~~ RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. 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.

needed to maintain sufficient coolant to the reactor vessel.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G 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.

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3.10.A (Cont'd)

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REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

The SRMs are provided to monitor the core during periods of ~~station~~ unit shutdown and to guide the operator during refueling operations and ~~station~~ unit startup. Requiring two OPERABLE SRMs (FLCs) ~~one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations.~~ Each SRM (FLC) is not required to read ≥ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

NEW ¶

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

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during CORE ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

REVISION TO TECHNICAL SPECIFICATION (TS) BASES (TS-348)
REVISED PAGES

I. AFFECTED PAGE LIST

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* Spillover pages

II. REVISED PAGES

See attached.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

F. (Deleted)

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I.J.& K. Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. Supplemental Reload Licensing Report of Browns Ferry Nuclear Plant, Unit 1 (applicable cycle-specific document).

2. GE Standard Application for Reactor Fuel, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved version).

3.2 <u>BASES</u> (Cont'd)

The low reactor water level instrumentation that is set to trip when reactor water level is 378 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the high steam flow instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

3.2 BASES (Cont'd)

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks or small breaks in the main steam lines. The trip setting of 200°F for the main steam line tunnel detector is low enough to provide early indication of a steam line break. Exceeding the trip setting causes closure of isolation valves. For large breaks, the high steam tunnel temperature detection instrumentation is a backup to the high steam flow instrumentation.

In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

3.5 **BASES** (Cont'd)

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is needed to maintain sufficient coolant to the reactor vessel. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. 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.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G 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.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were OPERABLE. By requiring six valves to be OPERABLE, additional conservatism is provided to account for the possibility of a single failure in the ADS system.

Reactor operation with one of the six ADS valves inoperable is allowed to continue for fourteen days provided the HPCI, core spray, and LPCI systems are OPERABLE. Operation with more than one ADS valve inoperable is not acceptable.

With one ADS valve known to be incapable of automatic operation, five valves remain OPERABLE to perform the ADS function. This condition is within the analyses for a small break LOCA and the peak clad temperature is well below the 10 CFR 50.46 limit. Analysis has shown that four valves are capable of depressurizing the reactor rapidly enough to maintain peak clad temperature within acceptable limits.

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 high point vent is visually checked for water flow once a month and 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 high point to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point 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 high point. The indicators will reflect approximately 30 psig for a water level at the high point 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' high points monthly.

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

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.

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 of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 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.

3.5.L. APRM Setpoints

The fuel cladding integrity safety limits of Section 2.1 were based on a total peaking factor within design limits (FRP/CMFLPD ≥ 1.0). The

APRM instruments must be adjusted to ensure that the core thermal limits are not exceeded in a degraded situation when entry conditions are less conservative than design assumptions.

3.5.M. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of Figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

During regional oscillations, the safety limit MCPR is not approached until APRM oscillations are 30 percent peak-to-peak or larger in magnitude. In addition, periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

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

5. Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request For Information On ODYN Computer Model," September 5, 1980.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

The SRMs are provided to monitor the core during periods of unit shutdown and to guide the operator during refueling operations and unit startup. Requiring two OPERABLE SRMs (FLCs) during CORE ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

Each SRM (FLC) is not required to read $\geq$ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

C. Spent Fuel Pool Water

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

The fuel pool cooling system is designed to maintain the pool water temperature less than 125°F during normal heat loads. If the reactor core is completely unloaded when the pool contains two previous discharge batches, the temperature may increase to greater than 125°F. The RHR system supplemental fuel pool cooling mode will be used under these conditions to maintain the pool temperature to less than 125°F.

3.10.D/4.10.D BASES

Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The five-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

3.10.E/4.10.E

Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

3.10.F Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety links provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety links are removed during decontamination.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all required refueling equipment interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its function.

B. Core Monitoring

Requiring the SRMs to be functionally tested prior to any CORE ALTERATION assures that the SRMs will be OPERABLE at the start of that alteration. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY.

REFERENCES

1. Fuel Pool Cooling and Cleanup System (BFNP FSAR Subsection 10.5)

2. Spent Fuel Storage (BFNP FSAR Subsection 10.3)

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the high steam flow instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks or small breaks in the main steam lines. The trip setting of 200°F for the main steam line tunnel detector is low enough to provide early indication of a steam line break. Exceeding the trip setting causes closure of isolation valves. For large breaks, the high steam tunnel temperature detection instrumentation is a backup to the high steam flow instrumentation.

3.2 BASES (Cont'd)

In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system. Each trip system consists of two elements. Each channel contains one temperature switch located in the pump room and three temperature switches located in the torus area. The RCIC high flow and high area temperature sensing instrument channels are arranged in the same manner as the HPCI system.

The HPCI high steam flow trip setting of 90 psid and the RCIC high steam flow trip setting of 450" H_2O have been selected such that the trip setting is high enough to prevent spurious tripping during pump startup but low enough to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

The HPCI and RCIC steam line space temperature switch trip settings are high enough to prevent spurious isolation due to normal temperature excursions in the vicinity of the steam supply piping. Additionally, these trip settings ensure that the primary containment isolation steam supply valves isolate a break within an acceptable time period to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

High temperature at the Reactor Water Cleanup (RWCU) System in the main steam valve vault, RWCU pump room 2A, RWCU pump room 2B, RWCU heat exchanger room or in the space near the pipe trench containing RWCU piping could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is needed to maintain sufficient coolant to the reactor vessel. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. 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.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G 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.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were operable. By requiring six valves to be OPERABLE, additional conservatism is provided to account for the possibility of a single failure in the ADS system.

Reactor operation with one of the six ADS valves inoperable is allowed to continue for fourteen days provided the HPCI, core spray, and LPCI systems are OPERABLE. Operation with more than one ADS valve inoperable is not acceptable.

With one ADS valve known to be incapable of automatic operation, five valves remain OPERABLE to perform the ADS function. This condition is within the analyses for a small break LOCA and the peak clad temperature is well below the 10 CFR 50.46 limit. Analysis has shown that four valves are capable of depressurizing the reactor rapidly enough to maintain peak clad temperature within acceptable limits.

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 high point vent is visually checked for water flow once a month and 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 high point to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point 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 high point. The indicators will reflect approximately 30 psig for a water level at the high point 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' high points monthly.

3.5 BASES (Cont'd)

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

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.

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 of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 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.

3.5.L. APRM Setpoints

Operation is constrained to the LHGR limit of Specification 3.5.J. 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. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

During regional oscillations, the safety limit MCPR is not approached until APRM oscillations are 30 percent peak-to-peak or larger in magnitude. In addition, periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. 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.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

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3.10 BASES (Cont'd)

3.10.A (Cont'd)

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

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Each SRM (FLC) is not required to read $\geq$ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

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1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

C. Spent Fuel Pool Water

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

The fuel pool cooling system is designed to maintain the pool water temperature less than 125°F during normal heat loads. If the reactor core is completely unloaded when the pool contains two previous discharge batches, the temperature may increase to greater than 125°F. The RHR system supplemental fuel pool cooling mode will be used under these conditions to maintain the pool temperature to less than 125°F.

D. Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The five-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

E. Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

The low reactor water level instrumentation that is set to trip when reactor water level is 378 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the ADS high drywell pressure bypass timer timed out, and the ADS timer timed out. In addition, at least one RHR pump or two core spray pumps must be running.

The ADS high drywell pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded for the worst case event. This temperature is well below the limiting PCT of 2200°F.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the high steam flow instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

3.2 <u>BASES</u> (Cont'd)

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks or small breaks in the main steam lines. The trip setting of 200°F for the main steam line tunnel detector is low enough to provide early indication of a steam line break. Exceeding the trip setting causes closure of isolation valves. For large breaks, the high steam tunnel temperature detection instrumentation is a backup to the high steam flow instrumentation.

In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" water for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

3.5 BASES (Cont'd)

3.5.F Reactor Core Isolation Cooling System (RCICS)

The RCICS functions to provide makeup water to the reactor vessel during shutdown and isolation from the main heat sink to supplement or replace the normal makeup sources. The RCICS provides its design flow between 150 psig and 1120 psig reactor pressure. Below 150 psig, RCICS is not required to be OPERABLE since this pressure is substantially below that for any events in which RCICS is needed to maintain sufficient coolant to the reactor vessel. RCICS will continue to operate below 150 psig at reduced flow until it automatically isolates at greater than or equal to 50 psig reactor steam pressure. 150 psig is also below the shutoff head of the CSS and RHRS, thus, considerable overlap exists with the cooling systems that provide core cooling at low reactor pressure. 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.

The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a COLD CONDITION. Steam pressure is sufficient at 150 psig to run the RCIC turbine for OPERABILITY testing, yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. Considering the low reactor pressure and the availability of the low pressure coolant systems during startup from a COLD CONDITION, twelve hours is allowed as a reasonable time to demonstrate RCIC OPERABILITY once sufficient steam pressure becomes available. The alternative to demonstrate RCIC OPERABILITY PRIOR TO STARTUP using auxiliary steam is provided for plant operating flexibility.

With the RCICS inoperable, a seven-day period to return the system to service is justified based on the availability of the HPCIS to cool the core and upon consideration that the average risk associated with failure of the RCICS to cool the core when required is not increased.

The surveillance requirements, which are based on industry codes and standards, provide adequate assurance that the RCICS will be OPERABLE when required.

3.5.G Automatic Depressurization System (ADS)

The ADS consists of six of the thirteen relief valves. It is designed to provide depressurization of the reactor coolant system during a small break loss of coolant accident (LOCA) if HPCI fails or is unable to maintain the required water level in the reactor vessel. ADS operation reduces the reactor vessel pressure to within the operating pressure range of the low pressure emergency core cooling systems (core spray and LPCI) so that they can operate to protect the fuel barrier. Specification 3.5.G 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.

The emergency core cooling system LOCA analyses for small line breaks assumed that four of the six ADS valves were OPERABLE. By requiring six valves to be OPERABLE, additional conservatism is provided to account for the possibility of a single failure in the ADS system.

Reactor operation with one of the six ADS valves inoperable is allowed to continue for fourteen days provided the HPCI, core spray, and LPCI systems are OPERABLE. Operation with more than one ADS valve inoperable is not acceptable.

With one ADS valve known to be incapable of automatic operation, five valves remain OPERABLE to perform the ADS function. This condition is within the analyses for a small break LOCA and the peak clad temperature is well below the 10 CFR 50.46 limit. Analysis has shown that four valves are capable of depressurizing the reactor rapidly enough to maintain peak clad temperature within acceptable limits.

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 high point vent is visually checked for water flow once a month and 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 high point to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point 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 high point. The indicators will reflect approximately 30 psig for a water level at the high point 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' high points monthly.

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

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.

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 of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 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.

3.5.L. APRM Setpoints

Operation is constrained to the LHGR limit of Specification 3.5.J. 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 one-percent plastic strain limit. A six-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. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of Figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

During regional oscillations, the safety limit MCPR is not approached until APRM oscillations are 30 percent peak-to-peak or larger in magnitude. In addition, periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A 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.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

3.10 **BASES** (Cont'd)

3.10.A (Cont'd)

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

The SRMs are provided to monitor the core during periods of unit shutdown and to guide the operator during refueling operations and unit startup. Requiring two OPERABLE SRMs (FLCs) during CORE ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

Each SRM (FLC) is not required to read $\geq$ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

C. Spent Fuel Pool Water

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

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3.10.D/4.10.D BASES

Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be OPERABLE for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The five-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

3.10.E/4.10.E

Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to

3.10 BASES (Cont'd)

attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

3.10.F Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety links provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety links are removed during decontamination.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all required refueling equipment interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its function.

B. Core Monitoring

Requiring the SRMs to be functionally tested prior to any CORE ALTERATION assures that the SRMs will be OPERABLE at the start of that alteration. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY.

REFERENCES

1. Fuel Pool Cooling and Cleanup System (BFNP FSAR Subsection 10.5)

2. Spent Fuel Storage (BFNP FSAR Subsection 10.3)

PS:JEM:SAW:MTN
Enclosures
cc (Enclosures):
R. R. Baron, BR 4J-C
E. S. Christenbury, ET 11H-K
K. N. Harris, LP 3B-C
R. W. Huston, Rockville Office-C
R. D. Machon, PAB 1E-BFN
T. J. McGrath, LP 3B-C
D. E. Nunn, LP 3B-C
T. W. Overlid, LP 4E-C
E. Preston, POB 2C-BFN
T. D. Shriver, PAB 1A-BFN
H. L. Williams, EDB 1A-BFN
RIMS, WT 3B-K

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