February 7, 1996

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1, 2, AND 3 REVISION TO TECHNICAL SPECIFICATION BASES (TS-370) (TAC NOS. M94158, M94159, AND M94160)

Dear Mr. Kingsley:

By letter dated November 17, 1995, the Tennessee Valley Authority (TVA) submitted changes to the Technical Specifications (TS) Bases for the Browns Ferry Nuclear Plant Units 1, 2, and 3. TVA is revising Units 1, 2, and 3 TS Bases to reflect the lighter weight of the GE11 fuel design and to provide consistency between Units 2 and 3 TS Bases.

The appropriate Units 1, 2, and 3 TS Bases modified pages are enclosed.

Please contact me at (301) 415-1470 if you have any questions regarding this topic.

Sincerely,

ORIGINAL SIGNED BY:

Joseph F. Williams, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

Enclosure: As stated

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REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 1

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*Overleaf page is provided to maintain document completeness.

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3.10/4.10.12	3.10/4.10.12

ENCLOSURE

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions.

1. All rods inserted
2. Refueling platform positioned near or over the core
3. Refueling platform main hoist is fuel loaded
4. Fuel grapple not full up
5. One rod withdrawn
6. * Refueling platform frame-mounted hoist is fuel loaded
7. * Refueling platform monorail hoist is fuel loaded
8. * Service platform hoist is fuel loaded

When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is

* The refueling platform frame-mounted, monorail and the service platform fuel-loaded hoist interlocks are required to be OPERABLE only when utilized for in-vessel fuel movements.

subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1,500 lbs, in comparison to the load-trip setting of 1,000 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400-lb load-trip setting on these hoists is adequate to trip the interlock when one of the more than 550-lb fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time without removing fuel from the cells. The maintenance is performed with the mode switch in the refuel position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated and that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that at least 0.38 percent Δk shutdown margin is available. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.7 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the refuel position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod.

Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. The requirements for SRM OPERABILITY during these CORE ALTERATIONS assure sufficient core monitoring.

REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 2

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*Overleaf pages are provided to maintain document completeness.

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1.1/2.1-8	1.1/2.1-8
1.1/2.1-9	1.1/2.1-9
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
1.1/2.1-14	1.1/2.1-14
1.1/2.1-15	1.1/2.1-15
1.1/2.1-16	1.1/2.1-16
1.1/2.1-16a	1.1/2.1-16a*
1.2/2.2-1	1.2/2.2-1*
1.2/2.2-2	1.2/2.2-2
1.2/2.2-3	1.2/2.2-3
1.2/2.2-4	1.2/2.2-4*
3.1/4.1-14	3.1/4.1-14
3.1/4.1-15	3.1/4.1-15
3.1/4.1-16	3.1/4.1-16*
3.1/4.1-17	3.1/4.1-17
3.1/4.1-18	3.1/4.1-18*
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3.1/4.1-21	3.1/4.1-21*
3.2/4.2-65	3.2/4.2-65
3.2/4.2-66	3.2/4.2-66*
3.2/4.2-67	3.2/4.2-67
3.2/4.2-68	3.2/4.2-68*
3.3/4.3-15	3.3/4.3-15
3.3/4.3-16	3.3/4.3-16*
3.3/4.3-17	3.3/4.3-17
3.3/4.3-18	3.3/4.3-18
3.3/4.3-19	3.3/4.3-19*
3.3/4.3-20	3.3/4.3-20
3.5/4.5-24	3.5/4.5-24
3.5/4.5-25	3.5/4.5-25*
3.5/4.5-30	3.5/4.5-30
3.5/4.5-31	3.5/4.5-31*
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3.5/4.5-33	3.5/4.5-33*

REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 2 (Continued)

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3.7/4.7-25	3.7/4.7-25*
3.7/4.7-26	3.7/4.7-26
3.7/4.7-27	3.7/4.7-27
3.7/4.7-28	3.7/4.7-28
3.9/4.9-19	3.9/4.9-19
3.9/4.9-20	3.9/4.9-20*
3.10/4.10-11	3.10/4.10-11*
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3.10/4.10-15	3.10/4.10-15
3.10/4.10-16	3.10/4.10-16*

1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

The fuel cladding represents one of the physical barriers which separate radioactive materials from environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system setpoints. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined in terms of the reactor operating conditions which can result in cladding perforation.

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the Fuel Cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). This establishes a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. Since boiling transition is not a directly observable parameter, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 2.1-1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in Specification 3.5.K) more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit 1.07 is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

1.1 BASES (Cont'd)

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.07 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1,100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1,400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50 percent. Thus, a core thermal power limit of 25 percent for reactor pressures below 800 psia is conservative.

For the fuel in the core during periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

2.1 BASES (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased High Flux Scram Trip Setting (RUN Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

During power increase transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased scram provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/HOT STANDBY MODE)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM System consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any SAFETY LIMIT is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is discussed in paragraph 7.5.5.4 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate and thus prevents scram actuation. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting is selected to provide adequate margin to the flow-biased scram setpoint.

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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. The scram feature that occurs when the main steam line isolation valves close shuts down the reactor 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 2 (applicable cycle-specific document).

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

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SAFETY LIMIT

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

Protective Action	Limiting Safety System Setting
A. Nuclear system relief valves open--nuclear system pressure	1,105 psig ± 11 psi (4 valves)
	1,115 psig ± 11 psi (4 valves)
	1,125 psig ± 11 psi (5 valves)
B. Scram--nuclear system high pressure	≤1,055 psig

1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are not high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressures (1,148 for suction and 1,326 for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients is added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the reload licensing submittal for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.2 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the pressure is not monitored at this point, it cannot be directly determined if this safety limit has been violated. Also, because of the potentially varying head level and flow pressure drops, an equivalent pressure cannot be a priori determined for a pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BFNP FSAR Sections 14.0 and Appendix N)

2. ASME Boiler and Pressure Vessel Code Section III

3. USAS Piping Code, Section B31.1

4. Reactor Vessel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)

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

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

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

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

3.1 BASES

The Reactor Protection System automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the LIMITING CONDITIONS FOR OPERATION necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The Reactor Protection System is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

3.1 BASES (Cont'd)

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL (with any control rod withdrawn from a core cell containing one or more fuel assemblies) and STARTUP Modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the STARTUP and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level, low scram pilot air header pressure and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

Because of the APRM downscale limit of ≥ 3 percent when in the RUN mode and high level limit of ≤15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power or a control rod scram will occur. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to SHUTDOWN). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

The low scram pilot air header pressure trip performs the same function as the high water level in the scram discharge instrument volume for fast fill events in which the high level instrument response time may be inadequate. A fast fill event is postulated for certain degraded control air events in which the scram outlet valves unseat enough to allow 5 gpm per drive leakage into the scram discharge volume but not enough to cause control rod insertion.

4.1 BASES

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP, or for which the only practical test is one that can be performed at SHUTDOWN.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

The once per six-month functional test frequency for the scram pilot air header low pressure trip function is acceptable due to:

1. The functional reliability previously demonstrated by these switches on Unit 2 during Cycles 6 and 7,

2. The need for minimizing the radiation exposure associated with the functional testing of these switches, and

3. The increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by more frequent functional testing.

4.1 BASES (Cont'd)

A single failure of one of the scram pilot air header low pressure trip switches would not result in the loss of the trip function. It is highly unlikely that two switches in one channel would experience an undetected failure during the period between six-month functional tests.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95 percent confidence level is proposed. With the (1-out-of-2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95 percent confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history.[1]

To facilitate the implementation of this technique, Figure 4.1-1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M = nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1-1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of one month will generally be used initially until a trend is established.

Group (B) devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purpose of analysis, it is assumed that this rare failure will be detected within two hours.

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1. Reliability of Engineered Safety Features as a Function of Testing Frequency, I. M. Jacobs, "Nuclear Safety," Vol. 9, No. 4, July-August, 1968, pp. 310-312.

The bistable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20×10^{-6} failure/hour. The bistable trip circuits are predicted to have unsafe failure rate of less than 2×10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above, and a weekly test interval for the bistable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bistable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1-1. There are numerous identical bistable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bistable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established at each refueling outage. There are several instruments which must be calibrated and it will take several hours to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRMs resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during the STARTUP/HOT STANDBY and REFUEL (with any control rod withdrawn from a core cell containing one or more fuel assemblies) Modes and inactive during full-power operation. Thus, the only test that is meaningful is the one performed prior to entering the applicable Mode (i.e., the tests that are performed prior to use of the instrument). Since testing of the IRM functions is not practical in the RUN Mode, testing is not required to be completed until 12 hours after entering the STARTUP/HOT STANDBY Mode from the RUN Mode. Twelve hours is based on operating experience and in consideration of providing reasonable time in which to complete the test.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of 0.4-percent would occur thus providing for adequate margin.

For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

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3.2 BASES

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

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

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

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

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

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

The low reactor water level instrumentation that is set to trip when reactor water level is $\geq$ 398 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.

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 channels. 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.

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.

The control rod block functions are provided to generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

When the RBM is required, the minimum instrument channel requirements apply. These requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are

2. The control rod housing support restricts the outward movement of a control rod to less than three inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts withdrawals and insertions of control rods to prespecified sequences. All patterns associated with these sequences have the characteristic that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in any pellet average enthalpy in excess of 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Reference Sections 3.6.6, 7.16.5.3, and 14.6.2 of the FSAR, and NEDE-24011-P-A, Amendment 17.

   In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10 percent of rated. Material in the cited reference shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 10 percent, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize individual control rod worth.

At power levels below 10 percent of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM constrains the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Reference Section 7.16.5.3 of the FSAR. The RWM functions as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. When the Rod Worth Minimizer is out of service, special criteria allow a second licensed operator or other technically qualified member of the plant staff to manually fulfill the control rod pattern conformance functions of this system. The requirement that the RWM be OPERABLE for the withdrawal of the first twelve rods on a startup is to ensure that a high degree of RWM availability is maintained.

The functions of the RWM make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 10 percent, the RWM forces adherence to acceptable (Banked Position Withdrawal Sequence or equivalent) rod patterns. Above 10 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 10 percent of rated power are imposed by power distribution requirements, as defined in Sections 3.5.I, 3.5.J, 4.5.I, and 4.5.J of these technical specifications.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One OPERABLE SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two OPERABLE SRMs are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2, and Dresden 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during STARTUP of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and STARTUP testing is sufficient to detect anomalous drive performance.

3. The 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10 percent of the control rods at 16-week intervals is adequate for determining the OPERABILITY of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWRs with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1 percent ΔK. Deviations in core reactivity greater than 1 percent ΔK are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of one percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

E. No BASES provided for this specification

F. Scram Discharge Volume

The nominal stroke time for the scram discharge volume vent and drain valves is $\leq$ 30 seconds following a scram. The purpose of these valves is to limit the quantity of reactor water discharged after a scram and no direct safety function is performed. The surveillance for the valves assures that system drainage is not impeded by a valve which fails to open and that the valves are OPERABLE and capable of closing upon a scram.

References

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

3.5 BASES

3.5.A. Core Spray System (CSS) and 3.5.B Residual Heat Removal System (RHRS)

Analyses presented in the FSAR* and analyses presented in conformance with 10 CFR 50, Appendix K, demonstrated that the core spray system in conjunction with two LPCI pumps provides adequate cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to below 2,200°F which assures that core geometry remains intact and to limit the core average clad metal-water reaction to less than 1 percent. Core spray distribution has been shown in tests of systems similar in design to BFNP to exceed the minimum requirements. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel.

The RHRS (LPCI mode) is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the core spray system; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI mode of the RHRS and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high-pressure emergency core cooling subsystems.

The intent of the CSS and RHRS specifications is to not allow startup from the cold condition without all associated equipment being OPERABLE. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis.

Should one core spray loop become inoperable, the remaining core spray loop, the RHR System, and the diesel generators are required to be OPERABLE should the need for core cooling arise. These provide extensive margin over the OPERABLE equipment needed for adequate core cooling. With due regard for this margin, the allowable repair time of seven days was chosen.

Should one RHR pump (LPCI mode) become inoperable, three RHR pumps (LPCI mode) and the core spray system are available. Since adequate core cooling is assured with this complement of ECCS, a seven day repair period is justified.

Should two RHR pumps (LPCI mode) become inoperable, there remains no reserve (redundant) capacity within the RHRS (LPCI mode). Therefore, the affected unit shall be placed in cold shutdown within 24 hours.

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*A detailed functional analysis is given in Section 6 of the BFNP FSAR.

Should one RHR pump (containment cooling mode) become inoperable, a complement of three full capacity containment heat removal systems is still available. Any two of the remaining pumps/heat exchanger combinations would provide more than adequate containment cooling for any abnormal or postaccident situation. Because of the availability of equipment in excess of normal redundancy requirements, a 30-day repair period is justified.

Should two RHR pumps (containment cooling mode) become inoperable, a full heat removal system is still available. The remaining pump/heat exchanger combinations would provide adequate containment cooling for any abnormal postaccident situation. Because of the availability of a full complement of heat removal equipment, a 7-day repair period is justified.

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment overpressurization. The containment cooling function of the RHRS is permitted only after the core has reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

Since the RHRS is filled with low quality water during power operation, it is planned that the system be filled with demineralized (condensate) water before using the shutdown cooling function of the RHR System. Since it is desirable to have the RHRS in service if a "pipe-break" type of accident should occur, it is permitted to be out of operation for only a restricted amount of time and when the system pressure is low. At least one-half of the containment cooling function must remain OPERABLE during this time period. Requiring two OPERABLE CSS pumps during cooldown allows for flushing the RHRS even if the shutdown were caused by inability to meet the CSS specifications (3.5.A) on a number of OPERABLE pumps.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core. Requiring two OPERABLE RHR pumps and one CSS pump provides redundancy to ensure makeup water availability.

Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

Since the RHR system cross-connect capability provides added long term redundancy to the other emergency and containment cooling systems, a

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.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.

3.5 BASES (Cont'd)

The LHGR shall be checked daily during reactor operation at $\geq$ 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 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.

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

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

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

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

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

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

REFERENCE

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

3.6.D/4.6.D Relief Valves

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

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

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

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

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

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

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

3. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973

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

3.6.E/4.6.E Jet Pumps

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

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

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

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

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

3.6.F/4.6.F Recirculation Pump Operation

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

Requiring at least one recirculation pump to be OPERABLE while in the RUN mode (i.e., requiring a manual scram if both recirculation pumps are tripped) provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

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

3.6.G/4.6.G Structural Integrity

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

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

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

Only proven nondestructive testing techniques will be used.

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

REFERENCES

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

3.7.A & 4.7.A Primary Containment

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Inerting

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

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

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

Vacuum Relief

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

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

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

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

3.9 BASES

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

The unit station-service transformer B for unit 1 or the unit station-service transformer B for unit 2 provide noninterruptible sources of offsite power from the 500-kV transmission system to the units 1 and 2 shutdown boards. Auxiliary power can also be supplied from the 161-kV transmission system through the common station-service transformers or through the cooling tower transformers by way of the bus tie board. The 4-kV bus tie board may remain out of service indefinitely provided one of the required offsite power sources is not supplied from the 161-kV system through the bus tie board.

The minimum fuel oil requirement of 35,280 gallons for each diesel generator fuel tank assembly is sufficient for seven days of full load operation of each diesel and is conservatively based on availability of a replenishment supply. Each diesel generator has its own independent 7-day fuel oil storage tank assembly.

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

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

The 480-V diesel auxiliary board may be out of service for short periods for tests and maintenance.

There is a safety related 250-V dc unit battery located in each unit. Each 250-V dc unit battery system consists of a battery, a battery charger, and a distribution panel. There is also a backup charger which can be assigned to any one of the three unit batteries. The 250-V dc unit battery systems provide power for unit control functions, unit DC motor loads and alternate control power to the 4160 and 480-V ac shutdown boards. The primary control power supplies to the 3A, 3C and 3D 4160-V ac shutdown boards and the Unit 3 480-V ac shutdown boards are also provided by unit batteries. There are five safety related 250-V dc shutdown battery systems assigned as primary control power supplies to

4160-V ac shutdown boards A, B, C, D, and 3EB. Each of these shutdown battery systems has as 250-V dc battery, a charger, and a distribution panel. A portable spare charger can be used to supply any one of the five shutdown battery systems.

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

The 250-V dc system is so arranged, and the batteries sized so that the loss of any one unit battery will not prevent the safe shutdown and cooldown of all three units in the event of the loss of offsite power and a design basis accident in any one unit. Loss of control power to any engineered safeguard control circuits is annunciated in the main control room of the unit affected. The loss of one 250-V shutdown board battery affects normal control power for the 480-V and 4,160-V shutdown boards which it supplies.

There are two 480-V ac RMOV boards that contain mg sets in their feeder lines. These 480-V ac RMOV boards have an automatic transfer from their normal to alternate power source (480-V ac shutdown boards). The mg sets act as electrical isolators to prevent a fault from propagating between electrical divisions due to an automatic transfer. The 480-V ac RMOV boards involved provide motive power to valves associated with the LPCI mode of the RHR system. Having an mg set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required. Since sufficient equipment is available to maintain the minimum complement required for RHR (LPCI) operation, a 7-day servicing period is justified. Having two mg sets out of service can considerably reduce equipment availability; therefore, the affected unit shall be placed in Cold Shutdown within 24 hours.

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

Specification 3.9.D provides the OPERABILITY requirements for emergency diesel generator power sources for the plant shared systems of standby gas treatment and control room emergency ventilation. This specification addresses the condition where one or more of the units is in cold shutdown, refueling, or is defueled, by requiring the diesel generators aligned to the shared systems to be OPERABLE when any of the BFN Units

3.10 BASES

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions.

1. All rods inserted
2. Refueling platform positioned near or over the core
3. Refueling platform main hoist is fuel loaded
4. Fuel grapple not full up
5. One rod withdrawn
6. * Refueling platform frame-mounted hoist is fuel loaded
7. * Refueling platform monorail hoist is fuel loaded
8. * Service platform hoist is fuel loaded

When the mode switch is in the REFUEL position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is

* The refueling platform frame-mounted, monorail and the service platform fuel-loaded hoist interlocks are required to be OPERABLE only when utilized for in-vessel fuel movements.

subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1,500 lbs, in comparison to the load-trip setting of 1,000 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400-lb load-trip setting on these hoists is adequate to trip the interlock when one of the more than 550-lb fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time without removing fuel from the cells. The maintenance is performed with the mode switch in the refuel position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated and that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that at least 0.38 percent Δk shutdown margin is available. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.7 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the REFUEL position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod.

Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. The requirements for SRM OPERABILITY during these CORE ALTERATIONS assure sufficient core monitoring.

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)

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REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 3

The revised pages contain marginal lines indicating the areas of change.
*Overleaf pages are provided to maintain document completeness.

Remove	Insert
1.1/2.1-8	1.1/2.1-8
1.1/2.1-9	1.1/2.1-9
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
1.1-2.1-14	1.1-2.1-14
1.1/2.1-15	1.1/2.1-15*
1.1/2.1-16	1.1/2.1-16
1.1/2.1-16a	1.1/2.1-17*
1.2/2.2-1	1.2/2.2-1*
1.2/2.2-2	1.2/2.2-2
1.2/2.2-3	1.2/2.2-3
1.2/2.2-4	1.2/2.2-4
3.1/4.1-15	3.1/4.1-15
3.1/4.1-16	3.1/4.1-16
3.1/4.1-19	3.1/4.1-19
3.1/4.1-20	3.1/4.1-20*
3.2/4.2-64	3.2/4.2-64
3.2/4.2-65	3.2/4.2-65*
3.2/4.2-66	3.2/4.2-66*
3.2/4.2-67	3.2/4.2-67
3.2/4.2-70	3.2/4.2-70
3.2/4.2-71	3.2/4.2-71
3.2/4.2-72	3.2/4.2-72
3.2/4.2-72a	3.2/4.2-72a*
3.3/4.3-13	3.3/4.3-13*
3.3/4.3-14	3.3/4.3-14
3.3/4.3-17	3.3/4.3-17
3.3/4.3-18	3.3/4.3-18
3.3/4.3-19	3.3/4.3-19*
3.3/4.3-20	3.3/4.3-20
3.5/4.5-27	3.5/4.5-27
3.5/4.5-28	3.5/4.5-28*
3.5/4.5-33	3.5/4.5-33
3.5/4.5-34	3.5/4.5-34*
3.6/4.6-30	3.6/4.6-30

REVISED TECHNICAL SPECIFICATION BASES PAGE LIST

Unit 3 (Continued)

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3.6/4.6-31	3.6/4.6-31
3.6/4.6-32	3.6/4.6-32
3.6/4.6-33	3.6/4.6-33
3.7/4.7-24	3.7/4.7-24*
3.7/4.7-25	3.7/4.7-25
3.7/4.7-26	3.7/4.7-26
3.7/4.7-27	3.7/4.7-27
3.7/4.7-34	3.7/4.7-34
3.7/4.7-35	3.7/4.7-35*
3.9/4.9-18	3.9/4.9-18
3.9/4.9-19	3.9/4.9-19*
3.10/4.10-11	3.10/4.10-11
3.10/4.10-12	3.10/4.10-12
3.10/4.10-13	3.10/4.10-13
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1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

The fuel cladding represents one of the physical barriers which separate radioactive materials from environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system setpoints. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined in terms of the reactor operating conditions which can result in cladding perforation.

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the Fuel Cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). This establishes a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. Since boiling transition is not a directly observable parameter, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 2.1-1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in Specification 3.5.K) more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit 1.07 is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

1.1 BASES (Cont'd)

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.07 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurrence of loss-of-cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1,100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1,400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of $28x10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50 percent. Thus, a core thermal power limit of 25 percent for reactor pressures below 800 psia is conservative.

For the fuel in the core during periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

2.1 BASES (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased High Flux Scram Trip Setting (RUN Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

During power increase transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased scram provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/HOT STANDBY MODE)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM System consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any SAFETY LIMIT is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is discussed in paragraph 7.5.5.4 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate and thus prevents scram actuation. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting is selected to provide adequate margin to the flow-biased scram setpoint.

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

2.1 BASES (Cont'd)

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. The scram feature that occurs when the main steam line isolation valves close shuts down the reactor 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 3 (applicable cycle-specific document).

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

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SAFETY LIMIT

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A.	Nuclear system relief valves open--nuclear system pressure	1,105 psig ± 11 psi (4 valves)
		1,115 psig ± 11 psi (4 valves)
		1,125 psig ± 11 psi (5 valves)
B.	Scram--nuclear system high pressure	≤1,055 psig

1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressures (1,148 for suction and 1,326 for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients is added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the reload licensing submittal for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.2 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the

1.2 BASES (Cont'd)

pressure is not monitored at this point, it cannot be directly determined if this safety limit has been violated. Also, because of the potentially varying head level and flow pressure drops, an equivalent pressure cannot be a priori determined for a pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BFNP FSAR Sections 14.0 and Appendix N)

2. ASME Boiler and Pressure Vessel Code Section III

3. USAS Piping Code, Section B31.1

4. Reactor Vessel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)

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

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

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

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

be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL (with any control rod withdrawn from a core cell containing one or more fuel assemblies) and STARTUP Modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the STARTUP and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level, low scram pilot air header pressure and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

Because of the APRM downscale limit of $\geq$ 3 percent when in the RUN mode and high level limit of $\leq$15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power or a control rod scram will occur. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to SHUTDOWN). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

The low scram pilot air header pressure trip performs the same function as the high water level in the scram discharge instrument volume for fast fill events in which the high level instrument response time may be inadequate. A fast fill event is postulated for certain degraded control air events in which the scram outlet valves unseat enough to allow 5 gpm per drive leakage into the scram discharge volume but not enough to cause control rod insertion.

4.1 BASES

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP, or for which the only practical test is one that can be performed at SHUTDOWN.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

The once per six-month functional test frequency for the scram pilot air header low pressure trip function is acceptable due to:

1. The functional reliability previously demonstrated by these switches on Unit 2 during Cycles 6 and 7,

2. The need for minimizing the radiation exposure associated with the functional testing of these switches, and

3. The increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by more frequent functional testing.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of 0.4-percent would occur thus providing for adequate margin.

For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

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3.2 BASES

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

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

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

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

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

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

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

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 channels. 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 3A, RWCU pump room 3B, 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.

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.

The control rod block functions are provided to generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

When the RBM is required, the minimum instrument channel requirements apply. These requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1-out-of-n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

Where: i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1 \times 10^{3}$$

$$= 40 \text{ days}$$

<u>For additional margin a test interval of once per month will be used initially.</u>

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure OPERABILITY of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but

(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time.

Forbidding simultaneous testing improves the availability of the system over that which would be achieved by testing each channel independently. These one-out-of-n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by Curve No. 1 of Figure 4.2-1 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^{3} hours.

If two similar channels are used in a 1-out-of-2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following, the second channel be bypassed, tested, and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

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3.3/4.3 BASES

A. Reactivity Limitation

1. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection 3.4 of the Final Safety Analysis Report, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38 percent Δk with the analytically determined strongest control rod fully withdrawn.

   The value of "R", in units of percent Δk, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

   The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local k_{∞}. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38 percent Δk. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity Margin - Inoperable Control Rods - Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a nonfully inserted position, that position shall be consistent with the shutdown reactivity limitations stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest OPERABLE control rod does not insert. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress-assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed rod after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings. The Rod Worth Minimizer is not automatically bypassed until reactor power is above the preset power level cutoff. Therefore, control rod movement is restricted and the single notch exercise surveillance test is only performed above this power level. The Rod Worth Minimizer prevents movement of out-of-sequence rods unless power is above the preset power level cutoff.

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the Rod Worth Minimizer.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2, and Dresder 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during STARTUP of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and STARTUP testing is sufficient to detect anomalous drive performance.

3. The 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10 percent of the control rods at 16-week intervals is adequate for determining the OPERABILITY of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWRs with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1 percent ΔK. Deviations in core reactivity greater than 1 percent ΔK are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of one percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

E. No BASES provided for this specification

F. Scram Discharge Volume

The nominal stroke time for the scram discharge volume vent and drain valves is $\leq$ 30 seconds following a scram. The purpose of these valves is to limit the quantity of reactor water discharged after a scram and no direct safety function is performed. The surveillance for the valves assures that system drainage is not impeded by a valve which fails to open and that the valves are OPERABLE and capable of closing upon a scram.

References

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

3.5 BASES

3.5.A. Core Spray System (CSS) and 3.5.B Residual Heat Removal System (RHRS)

Analyses presented in the FSAR* and analyses presented in conformance with 10 CFR 50, Appendix K, demonstrated that the core spray system in conjunction with two LPCI pumps provides adequate cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to below 2,200°F which assures that core geometry remains intact and to limit the core average clad metal-water reaction to less than 1 percent. Core spray distribution has been shown in tests of systems similar in design to BFNP to exceed the minimum requirements. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel.

The RHRS (LPCI mode) is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the core spray system; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI mode of the RHRS and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high-pressure emergency core cooling subsystems.

The intent of the CSS and RHRS specifications is to not allow startup from the cold condition without all associated equipment being OPERABLE. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis.

Should one core spray loop become inoperable, the remaining core spray loop, the RHR System, and the diesel generators are required to be OPERABLE should the need for core cooling arise. These provide extensive margin over the OPERABLE equipment needed for adequate core cooling. With due regard for this margin, the allowable repair time of seven days was chosen.

Should one RHR pump (LPCI mode) become inoperable, three RHR pumps (LPCI mode) and the core spray system are available. Since adequate core cooling is assured with this complement of ECCS, a seven day repair period is justified.

Should two RHR pumps (LPCI mode) become inoperable, there remains no reserve (redundant) capacity within the RHRS (LPCI mode). Therefore, the affected unit shall be placed in cold shutdown within 24 hours.

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*A detailed functional analysis is given in Section 6 of the BFNP FSAR.

Should one RHR pump (containment cooling mode) become inoperable, a complement of three full capacity containment heat removal systems is still available. Any two of the remaining pumps/heat exchanger combinations would provide more than adequate containment cooling for any abnormal or postaccident situation. Because of the availability of equipment in excess of normal redundancy requirements, a 30-day repair period is justified.

Should two RHR pumps (containment cooling mode) become inoperable, a full heat removal system is still available. The remaining pump/heat exchanger combinations would provide adequate containment cooling for any abnormal postaccident situation. Because of the availability of a full complement of heat removal equipment, a 7-day repair period is justified.

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment overpressurization. The containment cooling function of the RHRS is permitted only after the core has reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

Since the RHRS is filled with low quality water during power operation, it is planned that the system be filled with demineralized (condensate) water before using the shutdown cooling function of the RHR System. Since it is desirable to have the RHRS in service if a "pipe-break" type of accident should occur, it is permitted to be out of operation for only a restricted amount of time and when the system pressure is low. At least one-half of the containment cooling function must remain OPERABLE during this time period. Requiring two OPERABLE CSS pumps during cooldown allows for flushing the RHRS even if the shutdown were caused by inability to meet the CSS specifications (3.5.A) on a number of OPERABLE pumps.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core. Requiring two OPERABLE RHR pumps and one CSS pump provides redundancy to ensure makeup water availability.

Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

Since the RHR system cross-connect capability provides added long term redundancy to the other emergency and containment cooling systems, a

3.5 BASES (Cont'd)

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.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.

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

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

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

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

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

References

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

3.6.D/4.6.D Relief Valves

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

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

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

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

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

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

References

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

3. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973

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

3.6.E/4.6.E Jet Pumps

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

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

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is

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

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

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

3.6.F/4.6.F Recirculation Pump Operation

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

Requiring at least one recirculation pump to be OPERABLE while in the RUN mode (i.e., requiring a manual scram if both recirculation pumps are tripped) provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

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

3.6.G/4.6.G Structural Integrity

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

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

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

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

Only proven nondestructive testing techniques will be used.

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

References

1. BFNP FSAR Subsection 4.12, Inservice Inspection and Testing

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

3.7.A & 4.7.A Primary Containment

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

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

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

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

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

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

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

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

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

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

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

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

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

Inerting

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

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

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

Vacuum Relief

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

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

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

The valve position indicating lights consist of one check light on the check light panel which confirms full closure, one green light next to the hand switch which confirms 80° of full opening and one red light next to the hand switch which confirms "near closure" (within 3° of full closure). Each light is on a separate switch. If the check light circuit is OPERABLE when the valve is exercised by its air operator there exists a confirmation that the valve will fully close. If the red light circuit is OPERABLE, there exists a in the system, isolation is provided by high temperature in the cleanup system area. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

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

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

Group 7 - (Deleted)

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

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

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

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

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

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

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

3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to assist other sources of pressurization in maintaining the control room at a positive pressure.

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

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

3.9 BASES

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

The unit station-service transformer B for unit 3 provides a noninterruptible source of offsite power from the 500-kV transmission system to the unit 3 shutdown boards. Auxiliary power can also be supplied from the 161-kV transmission system through the common station-service transformers or through the cooling tower transformers by way of the bus tie board. The 4-kV bus tie board may remain out of service indefinitely provided one of the required offsite power sources is not supplied from the 161-kV system through the bus tie board.

The minimum fuel oil requirement of 35,280 gallons for each diesel generator fuel tank assembly is sufficient for seven days of full load operation of each diesel and is conservatively based on availability of a replenishment supply. Each diesel generator has its own independent 7-day fuel oil storage tank assembly.

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

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

The 480-V diesel auxiliary board may be out of service for short periods for tests and maintenance.

There is a safety related 250-V dc unit battery located in each unit. Each 250-V dc unit battery system consists of a battery, a battery charger, and a distribution panel. There is also a backup charger which can be assigned to any one of the three unit batteries. The 250-V dc unit battery systems provide power for unit control functions, unit DC motor loads and alternate control power to the 4160 and 480-V ac shutdown boards. The primary control power supplies to the 3A, 3C and

3D 4160-V ac shutdown boards and the Unit 3 480-V ac shutdown boards are also provided by unit batteries. There are five safety related 250-V dc shutdown battery systems assigned as primary control power supplies to 4160-V ac shutdown boards A, B, C, D, and 3EB. Each of these shutdown battery systems has as 250-V dc battery, a charger, and a distribution panel. A portable spare charger can be used to supply any one of the five shutdown battery systems.

The 250-V dc system is so arranged and the batteries sized so that the loss of any one unit battery will not prevent the safe shutdown and cooldown of all three units in the event of the loss of offsite power and a design basis accident in any one unit. Loss of control power to any engineered safeguard control circuits is annunciated in the main control room of the unit affected.

There are two 480-V ac RMOV boards that contain mg sets in their feeder lines. These 480-V ac RMOV boards have an automatic transfer from their normal to alternate power source (480-V ac shutdown boards). The mg sets act as electrical isolators to prevent a fault from propagating between electrical divisions due to an automatic transfer. The 480-V ac RMOV boards involved provide motive power to valves associated with the LPCI mode of the RHR system. Having an mg set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required. Since sufficient equipment is available to maintain the minimum complement required for RHR (LPCI) operation, a 7-day servicing period is justified. Having two mg sets out of service can considerably reduce equipment availability; therefore, the affected unit shall be placed in Cold Shutdown within 24 hours.

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

Specification 3.9.D provides the OPERABILITY requirements for emergency diesel generator power sources for the plant shared systems of standby gas treatment and control room emergency ventilation. This specification addresses the condition where one or more of the units is in cold shutdown, refueling, or is defueled, by requiring the diesel generators aligned to the shared systems to be OPERABLE when any of the BFN Units require OPERABILITY of the shared systems. The allowed out-of-service time of 30 days for the diesel generator aligned to the shared systems is commensurate with the importance of the affected systems when a unit is in cold shutdown, refueling, or is defueled; considers the low probability of a LOCA/Loss of offsite power in these conditions; and considers the availability of onsite power to redundant trains.

subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1,500 lbs, in comparison to the load-trip setting of 1,000 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400-lb load-trip setting on these hoists is adequate to trip the interlock when one of the more than 550-lb fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time without removing fuel from the cells. The maintenance is performed with the mode switch in the refuel position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated and that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that at least 0.38 percent Δk shutdown margin is available. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.7 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the REFUEL position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod.

Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. The requirements for SRM OPERABILITY during these CORE ALTERATIONS assure sufficient core monitoring.

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 temperatures 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.

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)

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