ENCLOSURE 1

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
BROWNS FERRY NUCLEAR PLANT
UNIT 1

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

B. Core Thermal Power Limit (Reactor Pressure ≤800 psia)

When the reactor pressure is less than or equal to 800 psia,

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR ≤ 13.4 kw/ft for 8X8, 8x8R, and P8x8R fuel, MCPR limits of Spec 3.5.k. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

2. APRM--When the reactor mode switch is in the STARTUP POSITION, the APRM scram shall be set at less than or equal to 15% of rated power.

3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

1.1 **BASES**

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 occurence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°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 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.
In addition to the boiling transition limit operation is constrained to a maximum LHGR of 13.4 kw/ft for all 8x8 fuels. This limit is reached when the Core Maximum Fraction of Limiting Power Density equals 1.0 (CMFLPD = 1.0). For the case where Core Maximum Fraction of Limiting Power Density exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated power and only with reduced APRM scram settings as required by specification 2.1.A.1.
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 powers and flow will always be greater than 4.56 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%. Thus, a core thermal power limit of 25% 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: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed throughout the spectrum of planned operating conditions up to the design thermal power condition of 3440 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 3293 MWt is the licensed maximum power level of Browns Ferry Nuclear Plant, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisions and results are summarized in References 1, 2, and 3.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications as further described in reference 4. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity has been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR > limits specified in specification 3.5.K is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

2.1 BASES

I. J. & K. Reactor low water level set point 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 set point and initiation set points. 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. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

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

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

4. Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC request for information on ODYN computer model," September 5, 1980.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

To meet the safety design basis, thirteen relief valves have been installed on the unit with a total capacity of 83.9% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steamline 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 1375 psig.

To meet the 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 valVe which is well below the allowable vessel overpressure of 1375 psig.

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential pressure	Indicator 0 to 2 psid	(1) (2) (5)

.B Control Rods

control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3. a. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Sequence Control System (RSCS) shall be operable except the RSCS constraints may be suspended by means of the individual rod bypass switches for
   1 - special criticality tests, or
   2 - control rod scram timing per 4.3.C.1.
   When RSCS is bypassed on individual rods for these exceptions RWM must be operable per 3.3.B.3.c and a second licensed operator may not be used in lieu of RWM.

4.3.B Control Rods

a. Verify that the control rod is following the drive by observing a response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the pre-set power level of the RSCS.

b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3.a. Prior to the start of control rod withdrawal at startup the capability of the Rod Sequence System (RSCS) to properly fulfill its functions shall be verified by the following checks:

Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences.

Group notch portion - For each of the six comparator circuits go through test initiate: comparator inhibit; verify; reset. On seventh attempt test is allowed to continue until completion is indicated by illumination of test complete light.

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3.3.B Control Rods

b. During the shutdown procedure no rod movement is permitted between the testing performed above 20% power and the reinstatement of the RSCS restraints at or above 20% power. Alignment of rod groups shall be accomplished prior to performing the tests.

c. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Worth Minimizer shall be operable A second licensed operator may verify that the operator at the reactor console is following the control rod program in lieu of RWM except as specified in 3.3.B.3.a.

d. If Specifications 3.3.B.3.a through .c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately.

4.3.B Control Rods

b. Prior to attaining 20% rated power during rod insertion at shutdown the tests in 4.3.B.3.a shall be performed to verify RSCS capability.

c. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:

1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified before reactor startup or shutdown.

2. The RWM computer on line diagnostic test shall be successfully performed.

3. Prior to startup, proper annunciation of the selection error of at least one out-of-sequence control rod shall be verified.

4. Prior to startup, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

5. Prior to obtaining 20% rated power during rod insertion at shutdown, verify the latching of the proper rod group and proper annunciation after insert errors.

d. When the RWM is not operable a second licensed operator will verify that the correct rod program is followed except as specified in 3.3.B.3.a.

3.3.B Control Rods

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

   a. Both RBM channels shall be operable:
   or

   b. Control rod withdrawal shall be blocked.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.500

4.3.B Control Rods

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and at least once per 24 hours thereafter.

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 800 psig This testing shall be completed prior to exceeding 40% power. Below 20% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. The sequence restraints imposed upon the control rods in the 100-50 percent rod density groups to the preset power level may be removed by use of the individual bypass switches associated with those control rods which are fully or partially withdrawn and are not within the 100-50 percent rod density groups. In order to bypass a rod, the actual rod axial position must be known; and the rod must be in the correct in-sequence position. As required by 3.3.B.3.a a second licensed operator may not be used in lieu of RWM for this testing.

3.3/4.3 BASES:

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to pre-specified 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. Ref. Sections 3.6.6, 7.7.A, 7.16.5.3, and 14.6.2 of the FSAR and NEDO-10527 and supplements thereto.

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20 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 20 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 20 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 and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide 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. Ref. Section 7.16.5.3 of the FSAR. They serve as a backup to procedure control of control rod sequences, which limit the maximum reactivity worth of control rods. Except during specified exceptions, when the Rod Worth Minimizer is out of service a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls to assure conformance.

* Because it is allowable by bypass certain rods in the RSCS during specified testing below 20 percent of rated power in the startup or run modes, a second licensed operator is not an acceptable substitute for the RWM during this testing.

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 SRM's 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 channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of 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.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (ie, MCPR given by Spec. 3.5.K or LHGR of 13.4 kw/ft. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

Scram Insertion Times

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

On an early BWR, some degradation of control rod scram performance occured during plant startup and was determined to be caused by

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. The CSS shall be operable:

(1) prior to reactor startup from a cold condition, or

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in specification 3.5.A.2.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing.

	Item	Frequency
a.	Simulated Automatic Actuation test	Once/ Operating Cycle
b.	Pump Operability	Once/ month
c.	Motor Operated Valve Operability	Once/ month
d.	System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a	Once/3 months

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

1. The RHRS shall be operable:

   (1) prior to a reactor startup from a Cold Condition; or

   (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in specifications 3.5.B.2, through 3.5.B.7

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are operable.

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain operable.

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

1. a.	Simulated Automatic Actuation Test	Once/Operating Cycle
b.	Pump Operability	Once/month
c.	Motor Operated valve operability	Once/month
d.	Pump Flow Rate	Once/3 months
e.	Test Check Valve	Once/Operating Cycle

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 15,000 gpm against an indicated system pressure of 200 psig.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

3. When it is determined that one RHR pump (LPCI mode) is inoperable at a time when operability is required, the remaining RHR pumps (LPCI mode) and active components in both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators shall be demonstrated to be operable immediately and daily thereafter.

3.5.I Average Planar Linear Heat Generation Rate

During steady state power operation, the Maximum Average Planar Heat Generation Rate (MAPHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1 through 3.5.I-5. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 Kw/ft.

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

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

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

The LHGR for 8x8, 8x8R, and P8x8R fuel shall be checked daily during reactor operation at ≥ 25% rated thermal power.

3.5.G Automatic Depressurization System

3. If specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be initiated and the reactor vessel pressure shall be reduced to 105 psig or less within 24 hours.

H. Maintenance of Filled Discharge Pipe

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

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

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

4.5.G Automatic Depressurization System

H. Maintenance of Filled Discharge Pipe

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

1. Every month prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

2. Following any period where the LPCI or core spray systems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. When the RHRS and the CSS are required to be operable, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

3.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) as a function of scram time and core flow, shall be equal to or greater than shown in Figure 3.5.K-1 multiplied by the K_f shown in Figure 3.5.2, where:

$\tau = 0$ or $\dfrac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}$, whichever is greater

τ_A=0.90 sec (Specification 3.3.C.1 scram time limit to 20% insertion from fully withdrawn)

$$\tau_B = 0.710 + 1.65 \left[\frac{N}{n}\right]^{\frac{1}{2}} (0.053)$$ [Ref 5]

$$\tau_{ave} = \frac{\sum_{i=1}^{n} \tau_i}{n}$$

n = number of surveillance rod tests performed to date in cycle (including BOC test).

τ_i= Scram time to 20% insertion from fully withdrawn of the i^{th} rod.

N = total number of active rods measured in specification 4.3.C.1 at BOC

If at any time during steady state operation it is determined by normal surveilance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

L. Reporting Requirements

If any of the limiting values identified in Specification 3.5.I, J, or K are exceeded during steady state operation and specified action is taken, the event shall be logged and reported in a 30-day report.

4.5.K. Minimum Critical Power Ratio (MCPR)

1. MCPR shall be determined daily during reactor power operation at>25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.

2. The MCPR limit shall be determined for each fuel type 8X8, 8X8R, P8X8R, from figure 3.5.K-1 respectively using:

   a. τ= 0.0 prior to initial scram time measurements for the cycle, performed in accordance with specification 4.3.C.1.

   b. τ as defined in specification 3.5.K following the conclusion of each scram time surveillance test required by specifications 4.3.C.1 and 4.3.C.2.

   The determination of the limit must be completed within 72 hours of each scram time surveillance required by specification 4.3.C.

3.5.J Linear Heat Generation Rate (LHGR)

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

The LHGR for 8x8, 8x8R, and P8x8R fuel shall be checked daily during reactor operation at $\geq$ 25% 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% rated thermal power, the MTPF would have to be greater than 10 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%, 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% 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. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values for MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specifications 3.5.I, .J, and .K, that if at any time during steady state power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving steady state operation beyond a specified limit shall be reported within 30 days. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative.

M. References

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

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

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

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

5. Letter from R. H. Buchholz (G.E.) to P. S. Check (NRC), "Response to NRC request for information on ODYN computer model," September 5, 1980.

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274L

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.9
10,000	12.1
15,000	12.2
20,000	12.1
25,000	11.6
30,000	10.9
35,000	9.9
40,000	9.3

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274H

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.1
1,000	11.2
5,000	11.8
10,000	12.1
15,000	12.2
20,000	12.0
25,000	11.5
30,000	10.9
35,000	10.0
40,000	9.3

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265L and P8DRB265L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.6
1,000	11.6
5,000	12.1
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

Table 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB284L, GLTA-1, GLTA-2

Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1000	11.3
5000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.2
30,000	10.8
35,000	10.2
40,000	9.5

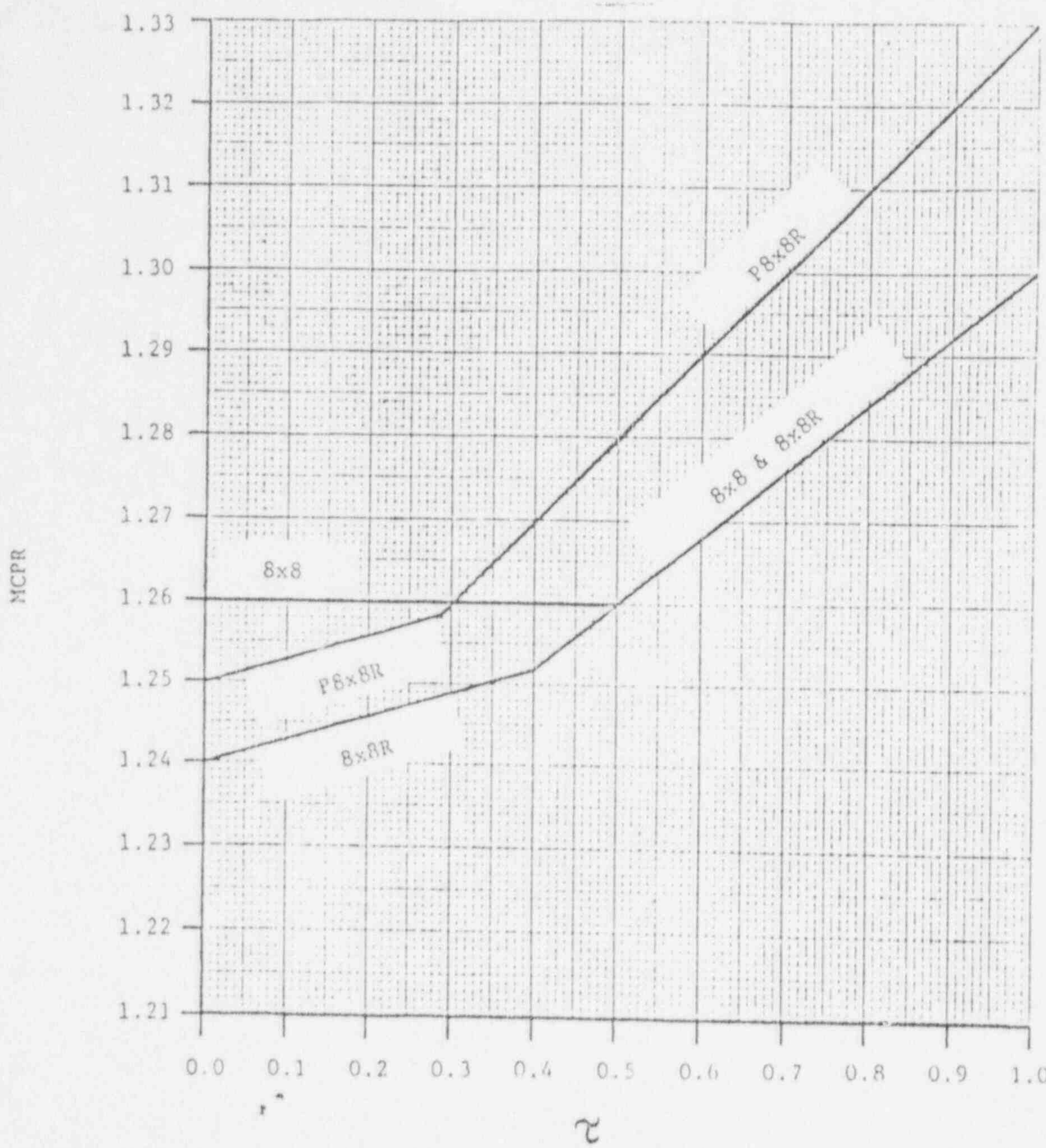

Figure 3.5.K-1
MCPR LIMITS*

*NOTE: Lead test assemblies are categorized as P8x8R bundles.

3.6.E Jet Pumps

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3. Steady state operation with both recirculation pumps out of service for up to 12 hrs is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor vessel water as determined by dome pressure.

C. Structural Integrity

1. The structural integrity of the primary system shall be

4.6.E Jet Pumps

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

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

2. Whenever there is recirculation flow with the reactor in the Startup or Run Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

4.6.F Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.

C. Structural Integrity

1. Table 4.6.A together with supplementary notes, specifies the

detected reasonably in a matter of 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 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)

3.6.D/4.6.D Relief Valves

To meet the safety basis thirteen relief valves have been installed on the unit with a total capacity of 83.9% 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 1375 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 1375 psig.

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

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

3.6.F/4.6.F Recirculation Pump Operation

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

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water level = -6.25" (differential pressure control >0 psid)

   -7.25" (0 psid differential pressure control)

   b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

   a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

3.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

      (1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

      (2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

   b. If the differential pressure of specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

3.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Whenever the reactor is not in cold shutdown, two independent gas analyzer systems shall be operable for monitoring the drywell and the torus.

2. With one hydrogen analyzer inoperable, restore at least two hydrogen analyzers to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 24 hours.

3. With no hydrogen analyzer OPERABLE the reactor shall be in HOT SHUTDOWN within 24 hours.

4.7 CONTAINMENT SYSTEMS

H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Each hydrogen analyzer system shall be demonstrated OPERABLE at least once per quarter by performing a CHANNEL CALIBRATION using standard gas samples containing a nominal eight volume percent hydrogen balance nitrogen.

2. Each hydrogen analyzer system shall be demonstrated OPERABLE by performing a CHANNEL FUNCTIONAL TEST monthly.

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
3	Reactor water cleanup system supply isolation valves FCV-69-1, & 2	1	1	30	0	GC
3	Reactor water cleanup system return isolation valves FCV-69-12		1	60	0	GC
4	HPCIS steamline isolation valves FCV-73-2 & 3	1	1	20	0	GC
5	RCICS steamline isolation valves FCV-71-2 & 3	1	1	15	0	GC
6	Drywell nitrogen purge inlet isolation valves (FCV-76-18)		1	5	C	SC
6	Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19)		1	5	C	SC
6	Drywell Main Exhaust isolation valves (FCV-64-29 and 30)		2	2.5	C	SC
6	Suppression chamber main exhaust isolation valves (FCV-64-32 and 33)		2	2.5	C	SC
6	Drywell/Suppression Chamber purge inlet (FCV-64-17)		1	2.5	C	SC
6	Drywell Atmosphere purge inlet (FCV-64-18)		1	2.5	C	SC

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	2.5	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	5	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	5	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	5	C	SC
6	Drywell/Suppression Chamber Nitrogen Purge Inlet (FCV-76-24)		1	5	C	SC
7	RCIC Steamline Drain (FCV-71-6A, 6B)		2	5	O	GC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	O	GC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-75-57, 58)		2	5	O	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement 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 at this time, 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 again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep offsite doses well below 10 CFR 100 limits.

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 released 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 specifications, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indication of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700 ft^3 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 3 feet and a water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will assure 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 majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

BASES

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operatibilit Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature of 170°F which is sufficient for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is not dependency on containment overpressure.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 200°F local. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressuirzed in a timely manner to avoid the regime of potentially high suppression chamber loadings.

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 assures margin for complete condensation of steam from the design basis loss-of-coolant accident.

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 loss-of-coolant accident 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-LOCA suppression pool hydrodynamic forces.

Inerting

The relatively 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 loss-of-coolant accident 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% hydrogen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

BASES

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable then the occurrence of the loss-of-coolant accident 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% following an accident, liquid nitrogen is maintained on-site for containment atmosphere dilution. About 2260 gallons would be sufficient as a 7-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2500 gallons is conservative. Following a loss of coolant accident the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed 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 system atmosphere as a second independent and redundant system will still be operable.

In terms of separability, redundancy for a failure of the torus system is based upon at least one operable drywell system. The drywell hydrogen concentration can be used to limit the torus hydrogen concentration during post LOCA conditions. Post LOCA calculations show that the CAD system initiated within two hours at a flow rate of 100 scfm will limit the peak drywell and wetwell hydrogen concentration to 3.6% (at 4 hours) and 3.8% (at 32 hours), respectively. This is based upon purge initiation after 20 hours at a flow rate of 100 scfm to maintain containment pressure below 30 psig. Thus, peak torus hydrogen concentration can be controlled below 4.0 percent using either the direct torus hydrogen monitoring system or the drywell hydrogen monitoring system with appropriate conservatism ($\leq$ 3.8%), as a guide for CAD/Purge operations.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 1 is located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The reactor core may contain 764 fuel assemblies consisting of 8x8 assemblies having 63 fuel rods each, and 8x8R (and P8x8R) assemblies having 62 fuel rods each.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of fuel in the new-fuel storage facility shall be such that k_{eff}, for dry conditions, is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

ENCLOSURE 2

EXPLANATION OF CHANGES

1. Incorporate changes associated with reload analysis, implementation of ODYN option B surveillance requirements, removal of all 7 x 7 fuel types, and elimination of 8 x 8 fuel power spiking penalty - pages vii, viii, 9, 16, 19, 25, 131, 159, 160, 169, 171, 172, 172a, 172b, 219, and 330.

2. Relax rod sequence control system requirements to allow low power physics testing - pages 122, 123, 124, and 129.

3. Clarification of requirements on charging ECCS systems - page 158

4. Mark I torus modifications; shorten downcomers and add T-quenchers - pages 227, 235a, 267, 268, and 269.

5. Containment purge modification to decrease valve closure time - pages 251 and 252.

6. Move natural circulation bases to appropriate section and rename section 3.6.F - pages 19, 182, and 221.

7. Miscellaneous corrections - pages 143, 145, and 173a.
8. Hydrogen Monitoring - pages 79, 249, and 270.

JUSTIFICATION

The justification provided consists of the GE report "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 1 Reload No. 4 (Cycle 5)" (Y1003J01A19) and Errata and Addenda to GE topical report "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 1 (NEDO-24056).

Justification for the new Hydrogen Monitoring System is provided. This justification is identical to that included with the unit 3 license amendment request TVA BFNP TS 148 (reference TVA letter from J.L.Cross to H.R.Denton dated September 5, 1980).

BROWNS FERRY NUCLEAR PLANT

PRIMARY CONTAINMENT HYDROGEN MONITORING SYSTEM

Each reactor is equipped with two totally independent systems for monitoring hydrogen concentrations in the drywell and the torus. Each system includes a Hays Republic Division, Milton Roy Company, Model 643D Condutherm thermal conductivity type gas analyzer, sample pumps, sample moisture removal equipment, and associated valves and controls, all mounted in a cabinet external to the primary containment. Gas samples are withdrawn from either the drywell or the torus for analysis.

The hydrogen analyzers use the principle of thermal conductivity to analyze the concentration of hydrogen in a gas mixture containing primarily nitrogen. The thermal conductivity of the sample changes linearly with the change in hydrogen concentration. The analyzer assembly consists of two identical electrically self-heated, glass-covered, temperature-sensitive resistors which are mounted in separate chambers in the analyzer cell block. These resistors form two legs of a Wheatstone bridge. A reference gas with known thermal conductivity diffuses into one of the cell chambers (the reference cell), and the drywell or torus sample diffuses into the other chamber (the measuring cell). The reference gas absorbs heat from the reference resistor in direct ratio to its thermal conductivity. The amount of heat absorbed remains constant since the reference gas has a constant thermal conductivity, maintaining the temperature and the resistance of the reference resistors constant. The sample gas absorbs heat from the measuring resistor in direct ratio to its thermal conductivity. As the composition of the sample gas changes, its thermal conductivity changes. This causes the amount of heat absorbed by the measuring resistor to change, changing the resistance of the measuring resistor. The Wheatstone bridge is calibrated so that 0 percent hydrogen balances the bridge, producing no electrical output. As the resistance of the measuring resistor changes, the bridge becomes unbalanced, producing an electrical output that is proportional to the hydrogen concentration in the gas sample. This signal is transmitted to a recorder in the control room.

Monitoring is continuous with an accuracy of 2 percent. No special operating procedures are required.

Each system is qualified for samples at 340° F, 45 psig, 100 percent RH, and post-LOCA fission product activity. All piping, cabling, the equipment cabinets, valves, readouts, and recorders are seismic Class I. Each system is powered from separate electrical fuses. All sample lines which penetrate the primary containment are equipped with one inboard and one outboard isolation valve per line.

Each system is designed to fully comply with the requirements of NUREG 0578 and Regulatory Guide 1.7 for primary containment hydrogen monitoring.